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UNRESOLVED SAFETY ISSUES SUMMARY

AQUA BOOK

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WASHINGTON, D.C. 20555

FOREWORD

THE "UNRESOLVED SAFETY ISSUES" SUMMARY IS DESIGNED TO PROVIDE THE MANAGEMENT OF THE NUCLEAR REGULATORY COMMISSION WITH A QUARTERLY OVERVIEW OF THE PROGRESS AND PLANS FOR COMPLETION OF GENERIC TASKS ADDRESSING "UNRESOLVED SAFETY ISSUES" REPORTED TO CONGRESS PURSUANT TO SECTION 210 OF THE ENERGY REORGANIZATION ACT OF 1974 AS AMENDED. THIS SUMMARY UTILIZES DATA COLLECTED FROM THE OFFICE OF NUCLEAR REACTOR REGULATION, OFFICE OF NUCLEAR REGULATORY RESEARCH, AND THE NATIONAL LABORATORIES AND IS PREPARED BY THE OFFICE OF MANAGEMENT AND PROGRAM ANALYSIS.

THE DEFINITION OF WHAT CONSTITUTES COMPLETION OF AN UNRESOLVED SAFETY ISSUE (USI) HAS RECENTLY BEEN EXPANDED TO INCLUDE THE IMPLEMENTATION OF THE TECHNICAL RESOLUTION. THIS IS IN ACKNOWLEDGEMENT OF THE FACT THAT REAL SAFETY BENEFITS OCCUR ONLY AFTER THE IMPLEMENTATION HAS TAKEN PLACE. IMPORTANT ELEMENTS OF THIS IMPLEMENTATION PHASE ARE:

1. THE PROVISION OF A PUBLIC COMMENT PERIOD FOLLOWING THE ISSUANCE OF A DRAFT NUREG REPORT INCORPORATING THE STAFF'S TECHNICAL RESOLUTION FOLLOWED BY A DISCUSSION AND DISPOSITION OF THE COMMENTS RECEIVED IN A FINAL NUREG REPORT.

2. THE PROVISION FOR INCORPORATION OF THE TECHNICAL RESOLUTION INTO THE NRC'S REGULATIONS, STANDARD REVIEW PLAN, REGULATORY GUIDES, OR OTHER NRC OFFICIAL GUIDANCE OR REQUIREMENTS AS APPROPRIATE.

3. THE PROVISION FOR APPLICATION OF THE TECHNICAL RESOLUTION TO INDIVIDUAL OPERATING PLANTS IN THE FORM OF HARDWARE OR DESIGN CHANGES, TECHNICAL SPECIFICATION CHANGE, AND/OR CHANGE TO OPERATING PROCEDURES AS APPROPRIATE.

THE SCHEDULES IN THIS BOOK INCLUDE A MILESTONE AT THE END OF EACH ACTION PLAN WHICH REPRESENTS THE INITIATION OF THE IMPLEMENTATION PROCESS BOTH WITH RESPECT TO INCORPORATION OF THE TECHNICAL RESOLUTION IN THE NRC OFFICIAL GUIDANCE OR REQUIREMENTS AND ALSO THE APPLICATION OF CHANGES TO INDIVIDUAL OPERATING PLANTS. THE SCHEDULE FOR IMPLEMENTATION WILL NOT NORMALLY BE INCLUDED IN THE TASK ACTION PLAN(S) FOR THE RESOLUTION OF A USI SINCE THE NATURE AND EXTENT OF THE ACTIVITIES NECESSARY TO ACCOMPLISH THE IMPLEMENTATION CANNOT NORMALLY BE REASONABLY DETERMINED PRIOR TO THE DETERMINATION OF A TECHNICAL RESOLUTION. THE PROGRESS AND STATUS FOR IMPLEMENTATION OF UNRESOLVED SAFETY ISSUES FOR WHICH A TECHNICAL RESOLUTION HAS BEEN COMPLETED ARE REPORTED SPECIFICALLY IN A SEPARATE TABLE PROVIDED IN THIS SUMMARY.

KARL KNIEL, CHIEF OF THE GENERIC ISSUES BRANCH, DIVISION OF SAFETY TECHNOLOGY/NRR, IS RESPONSIBLE FOR MANAGING THE GENERIC TASKS INCLUDED IN THIS SUMMARY.

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ABBREVIATIONS

AAB:	Accident Analysis Branch (Former NRR Branch)
AB:	Administration Branch, Training and Administration Staff (IE)
AC:	Alternating Current
ACRS:	Advisory Committee on Reactor Safeguards
AD:	Assistant Director
ADB:	Analysis and Development Branch, Division of Reactor Safety Research (RES)
AEB:	Accident Evaluation Branch, Division of Systems Integration (NRR)
AEOD:	Office of the Analysis and Evaluation of Operational Data
AIF:	Atomic Industrial Forum
APTS:	Action Plan Tracking System
ARL:	Alden Research Laboratory
ASB:	Auxiliary Systems Branch, Division of Systems Integration (NRR)
ASME:	American Society of Mechanical Engineers
ASTM:	American Society of Testing Materials
ATWS:	Anticipated Transient Without Scram
B&W:	Babcock and Wilcox Company
BNC:	Brookhaven National Conservatory
BNL:	Brookhaven National Laboratory
BOP:	Balance of Plant
BWR:	Boiling Water Reactor
CE:	Combustion Engineering, Incorporated
CEB:	Chemical Engineering Branch, Division of Engineering (NRR)
CFR:	Code of Federal Regulations
CO:	Condensation Isolation
CP:	Construction Permit
CPB:	Core Performance Branch, Division of Systems Integration (NRR)
CR:	Contractor Report
CRGR:	Committee for Review of Generic Requirements
CSB:	Containment Systems Branch, Division of Systems Integration (NRR)
DC:	Direct Current
DE:	Division of Engineering (NRR)
DFO:	Division of Facility Operations (RES)
DHFS:	Division of Human Factors Safety (NRR)
DHRS:	Decay Heat Removal System
DL:	Division of Licensing (NRR)
DOE:	U. S. Department of Energy
DOR:	Division of Operating Reactors (Former NRR Division)
DRA:	Division of Risk Analysis (RES)
DSI:	Division of Systems Integration (NRR)
DSS:	Division of Systems Safety (Former NRR Division)
DST:	Division of Safety Technology (NRR)
E:	Engineering
EB:	Enforcement Branch, Enforcement and Investigations Staff (IE)
ECC:	Emergency Core Cooling
EEB:	Environmental Engineering Branch, Division of Engineering (NRR)
EFPY:	Effective Full-Power Years
EG&G:	Edgerton, Germeshausen & Grier
EP:	Emergency Preparedness
EPRI:	Electric Power Research Institute
EQB:	Equipment Qualification Branch, Division of Engineering (NRR)
FIN:	Financial
FIRL:	Franklin Institute Research Laboratories
FSI:	Fluid Structure Interaction
FSTF:	Full-Scale Test Facility
FW:	Feedwater
GB:	Geosciences Branch, Division of Engineering (NRR)
GE:	General Electric
GIB:	Generic Issues Branch, Division of Safety Technology (NRR)
HFEB:	Human Factors Engineering Branch, Division of Health, Siting and Waste Mgmt. (RES)
HFS:	Division of Human Factors Safety (NRR)
HSST:	Heavy Section Steel Technology
ICSB:	Instrumentation and Control Systems Branch, Division of Systems Integration (NRR)
IE:	Office of Inspection and Enforcement
IEEE:	Institute of Electrical and Electronics Engineers
INEL:	Idaho Nuclear Engineering Laboratory
IREP:	Integrated Reliability Evaluation Program
ISI:	In-Service Inspection
LER:	Licensee Event Report
LGB:	Licensing Guidance Branch, Division of Safety Technology (NRR)
LLL:	Lawrence Livermore Laboratory
LOCA:	Loss-of-Coolant Accident
LPP:	Lead Plant Program
LTP:	Long Term Program
LWR:	Light-Water Reactor
MARK I-III:	Containment Types for Boiling Water Reactors
MEB:	Mechanical Engineering Branch, Division of Engineering (NRR)
MEBR:	Materials Engineering Branch, Division of Engineering Technology (RES)
MIT:	Massachusetts Institute of Technology
MMRB:	Metallurgy and Materials Branch, Division of Reactor Safety Research (RES)
MSLB:	Main Steam Line Break
MTEB:	Materials Engineering Branch, Division of Engineering (NRR)
NDE:	Non-Destructive Examination
NREP:	Neutron Resonance Escape Probability
NRL:	Naval Research Laboratory
NRR:	Office of Nuclear Reactor Regulation
NSS:	Nuclear Steam System
NUREG:	Nuclear Regulation
OG:	Owner's Group
OL:	Operating License

ABBREVIATIONS

ORAB:	Operating Reactors Assessment Branch, Division of Licensing (NRR)
ORB:	Operating Reactors Branch, Division of Licensing (NRR)
ORNL:	Oak Ridge National Laboratory
OSD:	Office of Standards Development (Former NRC Office)
OTSG:	Once-Through Steam Generator
PASNY:	Power Authority of the State of New York
PNL:	Pacific Northwest Laboratory (Battelle)
PS:	Plant Systems
PSB:	Power Systems Branch, Division of Systems Integration (NRR)
PSU:	Plant Systems Unit (AEOD)
PSYB:	Plant Systems Branch (Former NRR Branch)
PTRB:	Procedures and Test Review Branch, Division of Human Factors Safety (NRR)
PWR:	Pressurized Water Reactor
Q:	Quality Control or Assurances
RAB:	Radiological Assessment Branch, Division of Systems Integration (NRR)
RCIC:	Reactor Core Isolation Cooling
REF:	Reference
RES:	Office of Nuclear Regulatory Research
RFP:	Request for Proposal
RHR:	Residual Heat Removal
RPV:	Reactor Pressure Vessel
PRA:	Probability and Risk Assessment
RRAB:	Reliability and Risk Assessments Branch (NRR)
RRRC:	Regulatory Requirements Review Committee
RS:	Reactor Safety (Former NRR Branch)
RSB:	Reactor Systems Branch, Division of Systems Integration (NRR)
RSR:	Reactor Safety Research
RSSMAP:	Reactor Safety Study Methodology Application Program
SCC:	Stress-Corrosion Cracking
SEB:	Structural Engineering Branch, Division of Engineering (NRR)
SEP:	Systematic Evaluation Program
SEPB:	Systematic Evaluation Program Branch, Division of Licensing (NRR)
SER:	Safety Evaluation Report
SG:	Steam Generator
SGWH:	Steam Generator Water Hammer
SNL:	Sandia National Laboratory
SRP:	Standard Review Plan
SRV:	Safety Relief Valve
SSE:	Safe Shutdown Earthquake
SSPB:	Standards Special Projects Branch (NRR)
STP:	Short-Term Program
TAP:	Task Action Plan
TH:	Hot Leg Temperature
TM:	Task Manager
TMI:	Three Mile Island
USI:	Unresolved Safety Issue
W:	Westinghouse Electric Corporation
WH:	Water Hammer
WUTA:	Washington University Technology Associates

PROGRAM OVERVIEW

PROJECTED DATES FOR ISSUING NRC STAFF REPORTS

Key:
(1) LPP - Lead Plant Program
(2) LTP - Long Term Program
N/S - Not Scheduled

USI NO.	TITLE	NOV. 16, 1981 AQUA BOOK	CURRENT	REMARKS
A-1,	Water Hammer	January 15, 1983	June 1, 1983*	Work has resumed under contract to EG&G, Idaho. A design and procedures review will be conducted by Quadrex Corp. under subcontract to EG&G.
A-3, A-4, A-5,	Steam Generator Tube Integrity	March 31, 1982	February 1, 1983*	The draft NUREG report has been reviewed by the technical branches and their comments incorporated. The report is being prepared for management review and then will be issued for public comment.
A-7,	Mark I Long Term Program	Supplement-January 15, 1982	Supplement-March 31, 1982	This issue is considered complete since the final NUREG was issued July 1980 and confirmatory orders implementing the findings in NUREG 0661 were issued on January 13, 1981 for 22 operating BWRs with Mark I containments. However, BNL has reviewed three confirmatory issues for an SER supplement and issued a draft report for NRR review. The NRR supplement is scheduled to be issued 03-31-82.
A-11,	Reactor Vessel Materials Toughness	January 15, 1982	June 22, 1982*	NUREG-0774 Vols. I and II has been completed and was issued for public comment in September 1981. The comment period ended November 27, 1981. Many comments were received and most were resolved by editorial changes. The number of comments and the complexities of some will require more time to address than previously estimated.

*Part or all the schedule slippage for this task reflects additions to the milestone schedules to provide for review by the Committee for Review of Generic Requirements. (See individual schedules for details.)

PROGRAM OVERVIEW

PROJECTED DATES FOR ISSUING NRC STAFF REPORTS

USI NO.	TITLE	NOV. 16, 1981 AQUA BOOK	CURRENT	REMARKS
A-12,	Steam Generator & Reactor Coolant Pump Supports	January 15, 1982	July 19, 1982*	Meetings were held with EPRI and Industry groups in in December 1980, to resolve use of fracture mechanics methods for qualification of supports. NRC reviewer's comments on the final NUREG-0577 were not received until mid-January 1982, causing a delay in issuing the final report. Comments made by EPRI and the AIF subcommittee on material requirements will be incorporated into the final staff positions.
A-17,	Systems Interactions	See Item II.C.3 in Action Plan Tracking System	See Item II.C.3 in Action Plan Tracking System	Work planned under TAP A-17 will be performed under TMI Action Plan item II.C.3, Systems Interaction. The status of this item is reported in the quarterly updates of the TMI Action Plan Tracking System (APTS).
A-39,	SRV Pool Dynamic Loads	Mark II-Jan. 30, 1982 Mark III-Jan. 30, 1982	Mark II-June 15, 1982* Mark III-June 15, 1982*	NUREG-0763 on in-plant SRV tests was completed and issued in June 1981. NUREG-0783 on Pool temperature Limits was issued on October 26, 1981. NUREG-0802 on SRV Quencher loads has been prepared and is under management review.
A-40,	Seismic Design Criteria	N/S	December 31, 1982*	A draft report presenting staff position has been prepared and is under management review.
A-43,	Containment Emergency Sump	November 30, 1982	April 1, 1983*	The in-containment insulation survey is complete and NUREG/CR-2403 was issued in November 1981.

*Part or all the schedule slippage for this task reflects additions to the milestone schedules to provide for review by the Committee for Review of Generic Requirements. (See individual schedules for details.)

PROGRAM OVERVIEW
PROJECTED DATES FOR ISSUING NRC STAFF REPORTS

USI NO.	TITLE	NOV. 16, 1981 AQUA BOOK	CURRENT	REMARKS
A-44,	Station Blackout	March 31, 1983	June 30, 1983*	Technical Assistance Contracts with ORNL and Sandia have been issued. A request for information on Diesel Generator reliability was sent to all licensees in July 1981; responses from licensees are arriving.
A-45,	Shutdown Decay Requirements	October 1985	N/S	A Task Action Plan was issued on October 7, 1981. Funding for technical assistance on A-45 has not yet been approved and the Task Action Plan is being revised. A "HOLD" has been placed on this program.
A-46,	Seismic Qualification of Equipment in Operating Plants	December 15, 1983	N/S	This program is being redirected to ensure compatibility with the Equipment Qualification Program Plan.
A-47,	Safety Implications of Control Systems	N/S	N/S	A draft Task Action Plan has been prepared and is expected to be issued for staff concurrence by the end of February 1982.
A-48,	Hydrogen Control Measures and Hydrogen Burns on Safety Equipment	N/S	N/S	A Task Manager has been selected and the Task Action Plan is under development.
A-49,	Pressurized Thermal Shock	Did not appear in book - - - as of Nov. 16, 1981 A-49 was not classified as an USI.	N/S	A-49 was designated by the Commission as an USI on December 28, 1981. The Task Action Plan has been submitted for NRC management review and concurrence.

*Part or all the schedule slippage for this task reflects additions to the milestone schedules to provide for review by the Committee for Review of Generic Requirements. (See individual schedules for details.)

USI'S FOR WHICH TECHNICAL RESOLUTION IS COMPLETE

USI NO.	TITLE	DATE COMPLETED	REPORTS PUBLISHED	IMPLEMENTATION STATUS	IMPLEMENTATION TASK MANAGER/ TELEPHONE
A-2,	ASYMMETRIC BLOWDOWN LOADS ON REACTOR PRIMARY COOLANT SYSTEMS	DEC. 1980	NUREG-0609	ALL PWR PLANT ASSESSMENTS FOR ASYMMETRIC LOADS HAVE BEEN RECEIVED AND ARE CURRENTLY BEING EVALUATED BY STAFF AND CONTRACTOR EG&G. AS A BASIS FOR THIS EVALUATION, STAFF IS IMPLEMENTING CRITERIA DEVELOPED AS PART OF THE TASK ACTION PLAN A-2. THESE CRITERIA ARE INCLUDED IN NUREG-0609. B&W AND CE OWNERS' GROUPS HAVE RESPONDED TO THE QUESTIONS RAISED AS A RESULT OF OUR INITIAL REVIEW OF THEIR SUBMITTALS ON THE ASSESSMENT OF THE PRIMARY SYSTEM COMPONENTS AND SUPPORTS TO ASYMMETRIC LOCA LOADS. THESE RESPONSES ARE CURRENTLY BEING REVIEWED AND THE REVIEW IS EXPECTED TO BE COMPLETED BY NOVEMBER 1983. THE WESTINGHOUSE OWNERS GROUPS EXPECT A RESOLUTION OF THIS ISSUE ON THE BASIS OF 'LEAK BEFORE BREAK' CRITERIA. AN INTERNAL REPORT, RECOMMENDING ACCEPTANCE OF THE 'LEAK BEFORE BREAK' CONCEPT ISSUED BY THE STAFF IN FEBRUARY 1981, HAS BEEN DISCUSSED WITH THE ACRS AND IS UNDER REVIEW BY NRC MANAGEMENT.	JIM SHEA 492-7231
A-6,	MARK I SHORT TERM PROGRAM	DEC. 1977	NUREG-0408	COMPLETE - ALL PLANT UNIQUE ANALYSES AND EQUIPMENT MODIFICATIONS AS REQUIRED WERE REVIEWED AND ACCEPTED AND APPROPRIATE TECHNICAL SPECIFICATION CHANGES WERE MADE.	- - -

USI'S FOR WHICH TECHNICAL RESOLUTION IS COMPLETE

USI NO.	TITLE	DATE COMPLETED	REPORTS PUBLISHED	IMPLEMENTATION STATUS	IMPLEMENTATION TASK MANAGER/ TELEPHONE
A-7,	MARK I LONG TERM PROGRAM	JULY 1980	NUREG-0661	THE LICENSEES ARE IN THE PROCESS OF PERFORMING PLANT-UNIQUE ANALYSES AND DESIGNING AND INSTALLING MODIFICATIONS TO MEET THE COMMISSION'S ORDER DATE FOR EACH OPERATING PLANT. BASED ON THE STAFF'S REVIEW OF THE MARK I OWNERS GROUP JUNE 29, 1981 SUBMITTAL THE STAFF PREPARED A COMMISSION PAPER (SECY-81-678) REQUESTING COMMISSION APPROVAL FOR EXTENSION OF THE EXEMPTIONS AND COMPLETION DATES FOR PLANT MODIFICATIONS NEEDED TO COMPLY WITH THE STAFF'S ACCEPTANCE CRITERIA CONTAINED IN APPENDIX A TO NUREG-0661. COMMISSION APPROVAL WAS RECEIVED ON JANUARY 8, 1982. ORDERS EXTENDING THE EXEMPTIONS AND COMPLETION DATES WERE SENT OUT TO 21 OF THE 22 OPERATING REACTORS (COOPER NUCLEAR STATION WILL MEET THE EXISTING ORDER DATE AND THEREFORE DID NOT REQUIRE AN EXTENSION). THE LICENSEES ARE IN THE PROCESS OF PREPARING PLANT-UNIQUE ANALYSIS FOR STAFF REVIEW.	BYRON SIEGEL 492-7534
A-8,	MARK II CONTAINMENT POOL DYNAMIC LOADS	AUGUST 1981	NUREG-0808	THE REQUIREMENTS RECOMMENDED IN NUREG-0808 ARE BEING IMPLEMENTED DURING THE OPERATING LICENSE REVIEW FOR EACH PLANT WITH A MARK II CONTAINMENT. THESE REQUIREMENTS WILL ALSO BE INCLUDED AS AN ADDITION TO THE APPROPRIATE SECTION OF THE STANDARD REVIEW PLAN.	
A-9,	ATWS	SEPT. 1980	NUREG 0460, VOL. 4 46FR57521	TWO VERSIONS OF THE PROPOSED ATWS RULE HAVE BEEN ISSUED FOR COMMENT; THE STAFF VERSION AND AN ALTERNATE VERSION OF THE RULE PREPARED AT THE DIRECTION OF COMMISSIONER HENDRIE. FOLLOWING THE COMMENT PERIOD AND CONSIDERATION OF PUBLIC COMMENTS, A FINAL RULE WILL BE PREPARED FOR COMMISSION CONSIDERATION.	FRANK ROWSOME, RES 443-5913

USI'S FOR WHICH TECHNICAL RESOLUTION IS COMPLETE

USI NO.	TITLE	DATE COMPLETED	REPORTS PUBLISHED	IMPLEMENTATION STATUS	IMPLEMENTATION TASK MANAGER/ TELEPHONE
A-10,	BWR FEEDWATER NOZZLE CRACKING	NOV. 1980	NUREG-0619	RESPONSES FROM LICENSEES TO AN IMPLEMENTATION LETTER HAVE BEEN RECEIVED AND RECOMMENDED TREATMENT OF THESE RESPONSES HAVE BEEN SUBMITTED TO NRC MANAGEMENT. ADDITIONAL INFORMATION HAS BEEN REQUESTED OF LICENSEES. IMPLEMENTATION ACTION IS PROCEEDING. ELEVEN PLANTS HAVE RECEIVED LETTERS ACCEPTING THEIR PROPOSED MODIFICATION PLANS. ELEVEN PLANTS ARE BEING REVIEWED AND NEGOTIATIONS ARE UNDER WAY TO OBTAIN LICENSEE COMMITMENTS TO IMPLEMENT ALL NUREG-0619 GUIDANCE. LACROSSE, BIG ROCK POINT, AND DRESDEN 1 DO NOT HAVE SUSCEPTIBLE PLANT SYSTEM CONFIGURATIONS AND ARE CONSIDERED COMPLETE WITH REGARD TO THIS ACTION. HUMBOLDT BAY, BY VIRTUE OF ITS STATUS (SHUTDOWN, NO FORSEEABLE RESTART), IS ALSO CONSIDERED COMPLETE.	PAUL O'CONNOR 492-7215
A-24,	QUALIFICATION OF CLASS 1E SAFETY RELATED EQUIPMENT	AUG. 1981	NUREG-0588	THE NUREG REPORT WAS SUBMITTED TO THE PUBLISHER FOR PRINTING ON AUGUST 7, 1981. RESOLUTION OF PUBLIC COMMENTS AND ISSUANCE OF THE REVISED NUREG REPORT COMPLETES THE ACTIVITIES ON A-24. EQB HAS THE LEAD IN IMPLEMENTING THE POSITIONS IDENTIFIED IN THE REPORT.	MARK WILLIAMS 492-8285
A-26,	REACTOR VESSEL PRESSURE TRANSIENT PROTECTION	SEPT. 1978	NUREG-0224	ALL PLANTS WERE REQUESTED TO PROVIDE AN OVERPRESSURE PREVENTION SYSTEM THAT WOULD BE USED WHENEVER THE PLANT WAS IN A COLD SHUTDOWN CONDITION. ALL PWRS IMPLEMENTED THEIR SYSTEMS WITH PRELIMINARY APPROVAL FROM THE NRC, AND A COMPLETE REVIEW TOOK PLACE ON A POST-IMPLEMENTATION BASIS. THE REVIEW CONSISTS OF THREE PARTS: A) ELECTRICAL ASPECTS: B) REACTOR SYSTEM ASPECTS; C) TECHNICAL SPECIFICATIONS. PLANTS REMAINING TO BE EVALUATED ARE BEING REVIEWED BY CONTRACT WITH EG&G. THIS INCLUDES 9 OF THE 14 REMAINING PLANTS. FOR THE OTHER FIVE PLANTS, THE SER'S HAVE BEEN COMPLETED AND ARE IN THE PROCESS OF BEING ISSUED.	AL DEAGAZIO 492-8945

USI'S FOR WHICH TECHNICAL RESOLUTION IS COMPLETE

USI NO.	TITLE	DATE COMPLETED	REPORTS PUBLISHED	IMPLEMENTATION STATUS	IMPLEMENTATION TASK MANAGER/ TELEPHONE
A-31,	RESIDUAL HEAT REMOVAL REQUIREMENTS	1978	NO FORMAL REPORT SRP 5.4.7 REVISION	RRRC APPROVED IMPLEMENTATION PLAN JANUARY 31, 1978. IMPLEMENTATION FOR OPERATING REACTORS NOT COMPLETE.	TOM NOVAK 492-7817
A-36,	CONTROL OF HEAVY LOADS NEAR SPENT FUEL	JULY 1980	NUREG-0612	IMPLEMENTATION REQUIREMENTS WERE ISSUED TO ALL LICENSEES BY LETTER DATED DECEMBER 22, 1980. THE LETTER REQUESTED; INTERIM ACTIONS TO BE COMPLETED IN 90 DAYS, A PHASE I ACTION (REPORT, CONFIRMATION AND JUSTIFICATION) IN SIX MONTHS AND PHASE II (SPECIFIC REQUIREMENTS) IN NINE MONTHS. MOST LICENSEES HAVE RESPONDED TO THE DECEMBER 22, 1930 GENERIC LETTER AND THEIR RESPONSES ARE BEING EVALUATED. ABOUT NINE PLANTS HAVE REQUESTED EXTENSIONS TO THE SEPTEMBER 22, 1981 RESPONSE DATE. SEVERAL PLANTS HAVE NOT YET ADEQUATELY RESPONDED TO THE EXTENT THAT AN EVALUATION MAY BE COMPLETED.	BUD REQUA 492-7877
A-42,	PIPE CRACKS IN BOILING WATER REACTORS	JULY 1980	NUREG-0313 REV. 1	IN FEBRUARY 1981, NUREG-0313, REV. 1 WAS ISSUED TO ALL HOLDERS OF BWR OPERATING LICENSES OR CONSTRUCTION PERMITS AND TO ALL APPLICANTS FOR OPERATING LICENSES. BY JULY 1, 1981, THE APPLICANTS/LICENSEES WERE TO PROVIDE THEIR PROGRAM FOR REPLACEMENT OF SERVICE SENSITITIVE LINES AND WELDS, THEIR PROGRAM FOR AUGMENTED INSERVICE INSPECTION, THEIR PROGRAM FOR IMPROVING THE WATER CHEMISTRY ENVIRONMENT AND INCOPORATION OF ADEQUATE LEAK DETECTION CAPABILITY. ALL LICENSEES BUT ONE HAVE RESPONDED AND THEIR REPLIES ARE BEING EVALUATED BY A TECHNICAL SUPPORT CONTRACTOR (INEL). INITIAL TER'S ARE SCHEDULED TO BE COMPLETED IN THE FIRST QUARTER OF FY83.	DICK CLARK 492-7162

WATER HAMMER (A-1)

AS OF WEEK ENDING FEBRUARY 19, 1982

KEY PERSONNEL

TASK MANAGER
ALECK SERKIZ (DST) X24217

Alex. Serkiz 3/1/82

MPA ANALYST
B. DULIN X28330

TASK REVIEWERS

NAME	BRANCH
S. MCKAY	PTRB/DHFS
#C. GRAVES	RSB/DSI

SCHEDULED COMPLETION

1978 ANNUAL REPORT 12-31-80

CURRENT #06-01-83

• PROBLEM DESCRIPTION

Since 1971 there have been over 200 incidents involving water hammers in BWRs and PWRs reported. The water hammers (or steam hammers) have involved steam generator feedrings and piping, the RHR system, ECC systems, and containment spray, service water, feedwater and steam lines. The incidents have been attributed to such causes as rapid condensation of steam pockets, steam-driven slugs of water, pump startup with partially empty lines, and rapid valve motion. Most of the damage reported has been relatively minor, involving pipe hangers and restraints; however, there have been several incidents which have resulted in piping and valve damage.

Task Action Plan, TAP A-1, subdivides resolution of this Unresolved Safety Issue (USI) into the following tasks:

Task 1: Water Hammer Summary Reports

Task 2: Resolution of Water Hammer Issue

Task 3: Water Hammer Position for Operating Reactors

Task 4: Water Hammer Safety Studies

Task 1 and 4 have been completed and the reports are on the network charts which follow. The near-term position for operating reactors is to permit continued operation since, as noted above, damage attributed to water hammer has been relatively minor. The long-term aspect of Task 3, is an integral part of the resolution of this issue.

Task 2 represents the resolution aspect of this USI and will result in the final resolution.

• RES INTERFACE INFORMATION

None.

• ACRS INTERFACE INFORMATION

A presentation was made to the ECCS Subcommittee on 12/03/81, plans for resolution of A-1 were presented.

• TECHNICAL ASSISTANCE CONTRACTS

Resolution efforts for A-1 are being conducted under FIN No. A6451 at INEL. Both EG&G-Idaho and Quadrex staff are being utilized.

FIN NO.	CONTRACTOR		OBLIGATED	EXPENDED
FY 81: A6451	EG&G-Idaho		$120,000	$ 39,355
FY 82: A6451	EG&G-Idaho	10/81	$120,000	$ 44,069
		11/81	145,000	56,265
		12/81	305,000	67,610 Est.
		01/82	305,000	137,195 Est.

• POTENTIAL PROBLEMS

Incorporation of estimated CRGR review actions into the projected A-1 resolution schedule leads to an estimated 4 month slippage (see also revised Task 2 milestones).

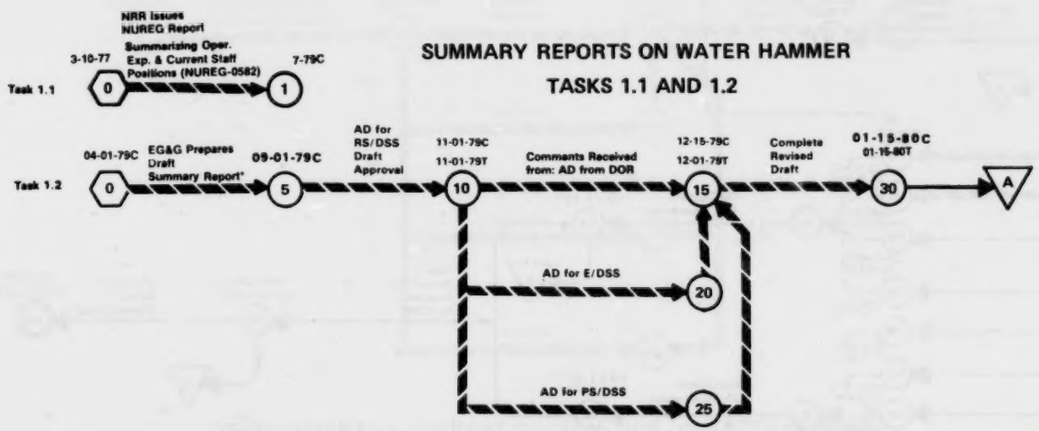
• STATUS SUMMARY

The concluding efforts and scheduled milestones for resolution of A-1 are shown in the attached schedule. Resolution will be based on evaluations of plant systems and operating procedures, severity of events experienced and attendant safety implications. It is postulated that guidelines and procedures will be developed which will minimize, or mitigate, water hammer events. The frequency of water hammer events reported has been low, and the incurred damage principally confined to pipe hangers and snubbers.

A final draft of the water hammer experience reports was received 01/10/82; review comments were provided to EG&G and final revisions are underway. A 03/30/82 estimated completion date (camera-ready copy) is now projected—see Milestone #112.

Quadrex (under sub-contract to EG&G-Idaho) has completed a preliminary assessment of water hammer in PWR feedwater systems and BWR core spray systems; the results are being reviewed. A preliminary assessment of the BWR RHR systems is currently underway.

WATER HAMMER (A-1)



TASK 1 REPORTS

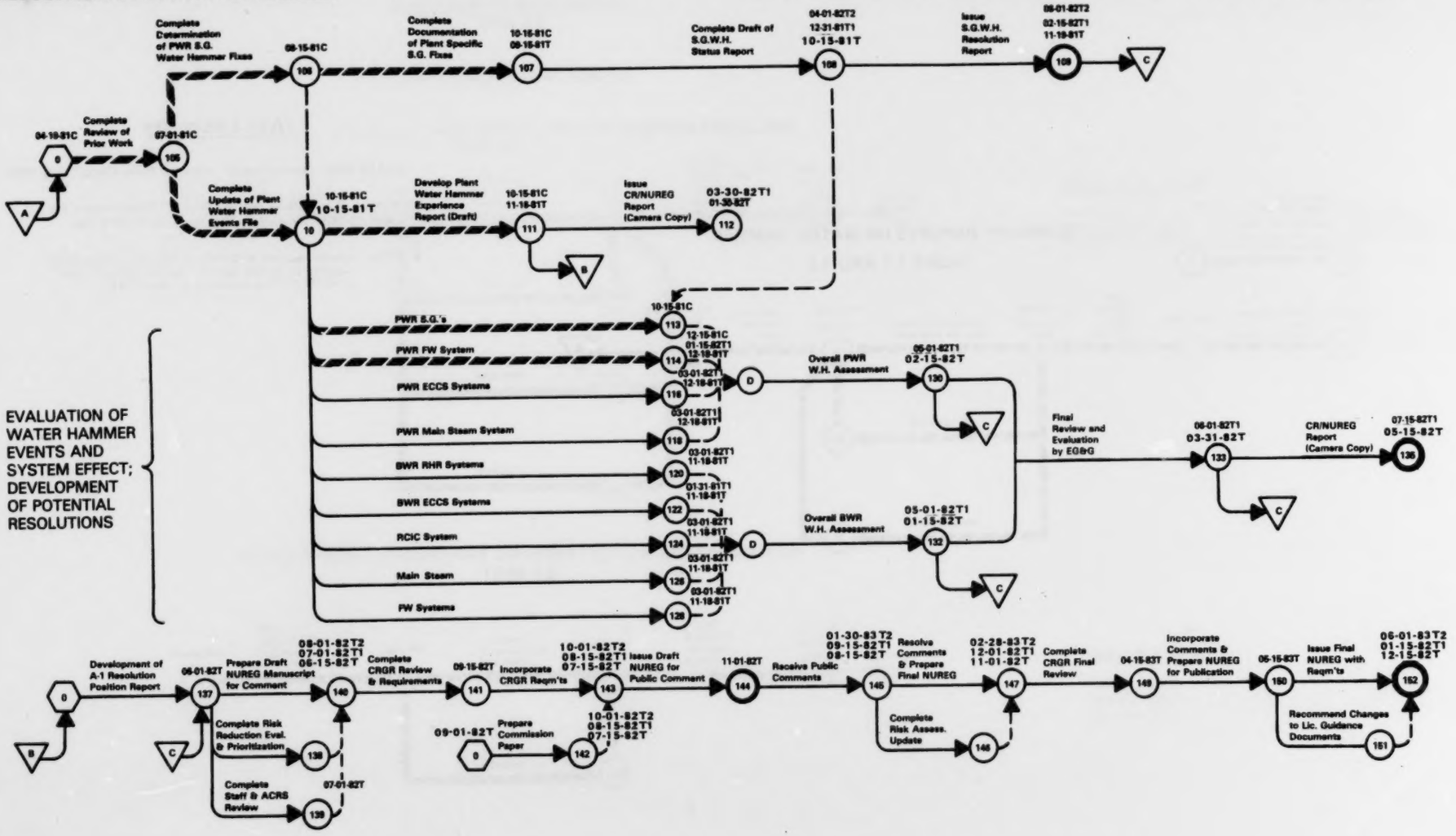
NUREG-0582, "Water Hammer in Nuclear Power Plants," July 1979

Chapman, R.L., "Water Hammer Studies," EG&G Interim Rpt., CAAP-TR-053 (Rev. 1), July 1980 (Prepared for NRC Use)

NOTE: Events reported in CAAP-TR-053 (Rev. 1) are being updated to include occurrences thru mid CY 1981. A NUREG/CR is scheduled for April 1982.

TASK 2.0
RESOLUTION OF A-1, WATER HAMMER

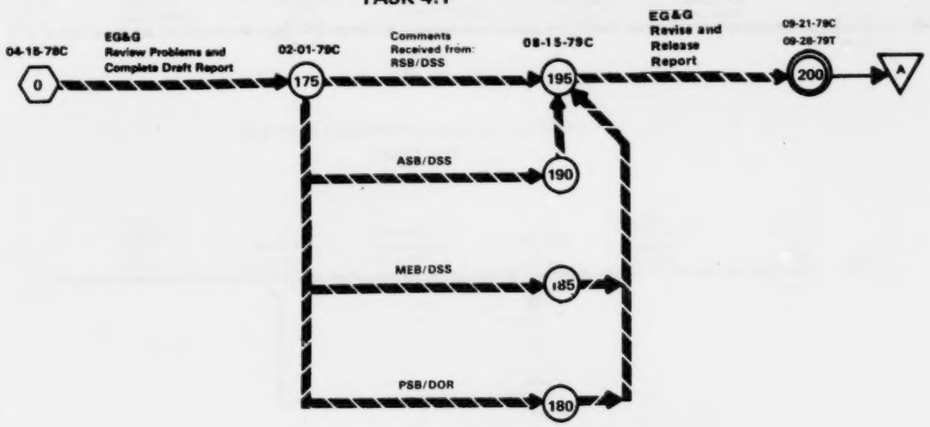
WATER HAMMER (A-1) (Continued)



DOR LONG-TERM POSITION
TASK 3.2

INCORPORATED IN SUBTASK 2.0

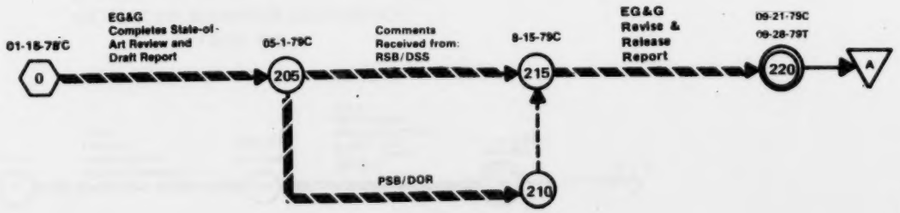
REVIEW AND EVALUATION OF POTENTIAL WATER HAMMER PROBLEMS
TASK 4.1



TASK 4.1 REPORT

Chapman, R. L., et al., "Review and Evaluation of Actual and Potential Water Hammer Events in Nuclear Plants," EG&G Interim Rpt, CAAP-TR-042 (Rev. 1), September 1979 (Prepared for NRC Use)

DEVELOPMENT OF CURRENT INFORMATION ON WATER HAMMER
TASK 4.2

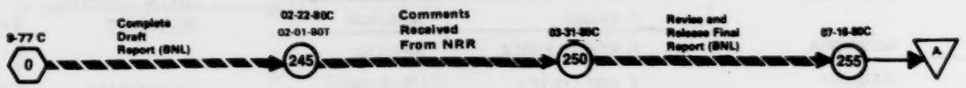


TASK 4.2 REPORT

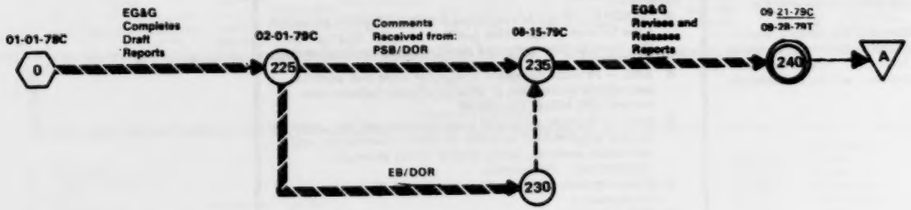
Watkins, J.C., Berry, R.A., "A State-of-the-Art Literature Review of Water Hammer," EG&G Interm Interim Rpt., RE-A-70-044, April 1979 (Prepared for NRC Use)

WATER HAMMER (A-1)
Continued

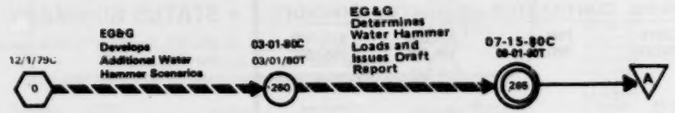
WATER HAMMER IN STEAM GENERATORS
TASK 4.3



WATER HAMMER CALCULATIONS
TASK 4.4



EVALUATE SAFETY SIGNIFICANCE OF WATER HAMMER SCENARIOS
TASK 4.5



TASK 4.3 REPORT

Saha, P., et al., "An Evaluation of Condensation - Induced Water Hammer in Preheat Steam Generators," NUREG/CR-1808, September 1980

TASK 4.4 REPORTS

Williamson, R.L., "An Analysis Tool for Predicting the Transient Hydrodynamics Resulting from the Rapid Filling of Voided Piping Systems," EG&G Interim Rpt, RE-E-79-008, February 1979 (Prepared for NRC Internal Use)

Morton, D.K., "An Analytical Procedure for Performing Structural Analyses of Nuclear Piping Sytems Subjected to Fluid Transients," EG&G Interim Rpt, RE-E-79-013, February (Prepared for NRC Internal Use)

Dommie, P.N., "An Investigation of the Steam Void Collapse Water Hammer Initiating Mechanism," EG&G Interim Rpt., RE-A-79-229, February 1979 (Prepared for NRC Internal Use)

Berry, R.A., "An Analysis Tool for Predicting Transient Hydrodynamics in Nuclear Piping Systems Containing Swing Check Valves," EG&G Rpt, RE-A-79-291 (Rev. 2), September 1979 (Prepared for NRC Internal Use)

TASK 4.5 REPORT

Saffell, B.F., "Supplemental Water Hammer Analysis and System Review, "EG&G preliminary Rpt, EGG-CAAP-5133, July 1980 (Prepared for NRC for Use as a Preliminary or Working Document)

WEST., CE & B&W STEAM GENERATOR TUBE INTEGRITY (A-3, A-4, A-5)

AS OF WEEK ENDING FEBRUARY 19, 1982

KEY PERSONNEL

TASK MANAGER
J. STROSNIDER X36903

P. Norian for J. Strosnider

MPA ANALYST
B. DULIN X28330

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F. ODAR	ADB/RSR
F. AKSTULEWICZ	AEB/DSI
J. ZWOLINSKI	HFEB/DHFS
L. FRANK	MEBR

SCHEDULED COMPLETION

1978 ANNUAL REPORT — Early 1980

CURRENT — # 02-01-83

• PROBLEM DESCRIPTION

Pressurized water reactor steam generator tube integrity can be degraded by corrosion induced wastage, cracking, reduction in tube diameter (denting) and vibration induced fatigue cracks. The primary concern is the capability of degraded tubes to maintain their integrity during normal operation and under accident conditions (LOCA or a main steam line break) with adequate safety margins.

Westinghouse and Combustion Engineering steam generator tubes have suffered degradation due to wastage and stress corrosion cracking. Both types of degradation have been decreased by conversion from phosphate to an all-volatile secondary water treatment. Degradation due to denting which leads to primary side stress corrosion cracking continues to be a problem.

B&W's once-through steam generators (OTSG's) were relatively free of trouble prior to the first tube leak incident at Oconee Unit 3 in July, 1976. Since then, all three Oconee units have experienced tube leak incidents. The leaks at the Oconee units are the result of cracks of unknown origins propagated in the circumferential direction by flow induced vibration and have been limited to tubes located adjacent to the open tube inspection lane.

A second form of degradation characterized as an erosion-cavitation phenomena has been observed at Oconee and other B&W units.

• RES INTERFACE INFORMATION

A. RES has funded, at the request of NRR, a major confirmatory program at PNL. The activity of this program consists of tests to verify the burst and cyclic strengths of degraded steam generator tubes and the leak rate data.
B. RES is funding a program addressing the factors which determine Inconel 600 susceptability to stress corrosion cracking in primary water. Metal condition, chemistry, temperature, stress and environment will be considered.

• ACRS INTERFACE INFORMATION

NONE

• TECHNICAL ASSISTANCE CONTRACTS

The following technical assistance contracts are generic in nature and will be applicable to the three Category "A" Technical Activities (A-3, A-4, and A-5) related to PWR steam generators.

A SANDIA — Provide statistical analysis of steam generator tube failures in operating reactors in order to establish the bases for the sampling plan for inservice inspection. Completed.
B BNL — Provide necessary computer code and perform parametric evaluation of effects of tube failures concurrent with MSLB. Completed
C BNL — Provide technical consultation and assistance to review information in areas of water chemistory and corrosion analysis, stress and/or burst strength. calculations. Completed.
D PNL — Provide cost/benefit evaluation of ISI plans. Completed.
E PNL — Evaluate environmental consequences of multiple tube failures concurrent with MSLB. Completed.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
2314	PNL	$75,000	$75,000
B2315	PNL	$95,000	$95,000

• POTENTIAL PROBLEMS

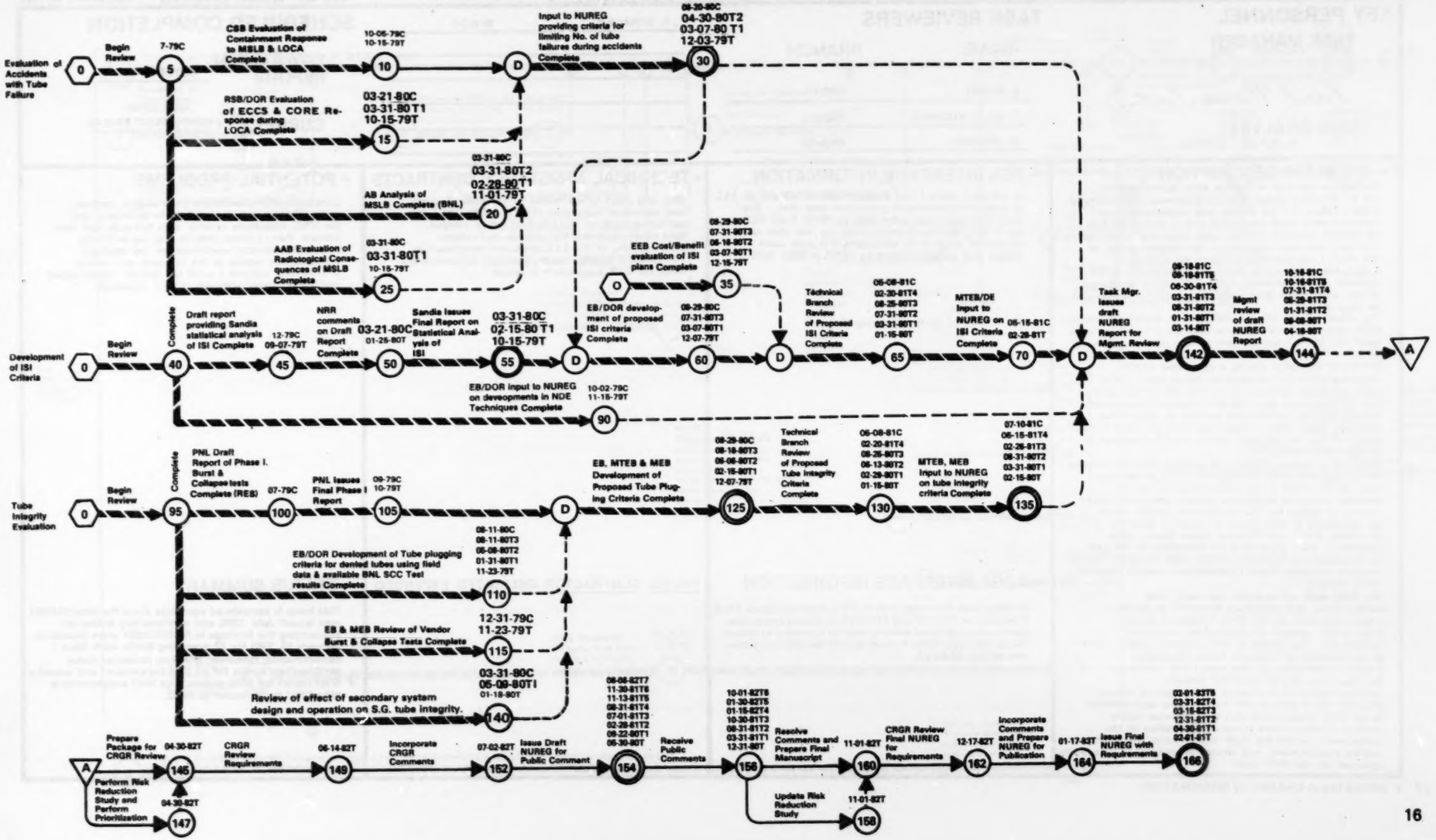
Management review and concurrence and CRGR approval are the only remaining actions necessary for issuance of the "For Comment" NUREG report.

• STATUS SUMMARY

The "For Comment" NUREG report is being prepared for submittal to CRGR.

# INDICATES A CHANGE IN INFORMATION

WEST., CE & B&W STEAM GENERATOR TUBE INTEGRITY (A-3, A-4,& A-5)



MARK I CONTAINMENT LONG-TERM PROGRAM (A-7)

AS OF WEEK ENDING FEBRUARY 19, 1982

KEY PERSONNEL

TASK MANAGER

Paul Norian (acting)

MPA ANALYST
B. DULIN X28330

TASK REVIEWERS

NAME	BRANCH
B. SIEGEL	ORB/DL
T. SU, J. KUDRICK	CSB/DSI
R. BOSNAK	MEB/DE
F. SCHAUER	SEB/DE

SCHEDULED COMPLETION

1978 ANNUAL REPORT	OCTOBER 1979
CURRENT	JULY 1980C # SUPPLEMENT 03-31-82

• PROBLEM DESCRIPTION

During the conduct of a large scale testing program for an advanced design pressure-suppression containment system (Mark III) for BWRs, new suppression pool hydrodynamic loads associated with a postulated loss of coolant accident (LOCA) were identified which had not been explicitly included in the original design of the Mark I containment systems. In addition, experience at operating plants has indicated that the dynamic effects of safety-relief valve (SRV) discharges to the suppression pool could be substantial and should be reconsidered.

The results of the Mark I containment short-term program (STP) have provided assurance that for a "most probable load" the Mark I containment system of each operating BWR facility would maintain its integrity and functional capability during a postulated LOCA.

The need exists (1) to establish design basis LOCA loads which are appropriate for the life of the facility, and (2) to restore the originally-intended design safety margins for the containment systems or ensure that adequate design safety margin has been provided in the design of the containment system prior to issuance of an operating license.

The Mark I Owner's Group has initiated a comprehensive testing and evaluation program to define design basis loads for the Mark I containment system and to establish structural acceptance criteria which will assure margins of safety for the containment system which are equivalent to that which is currently specified in the ASME Boiler and Pressure Vessel Code. Also included in their program is an evaluation of the need for structural modifications and/or load mitigation devices to assure adequate Mark I containment system structural safety margins.

The NRC staff will evaluate the loads, load combinations, and associated structural acceptance criteria proposed by the Mark I Owner's Group prior to the conduct of plant-unique structural evaluations. The results of this evaluation will be documented in a generic Safety Evaluation Report. Publication of this report will constitute the resolution of this Technical Activity.

Implementation of the results of this generic review, although not a part of this task, will be accomplished by an NRC requirement that each affected utility perform a plant-unique structural evaluation of the containment system for their facility using the loads, loading combinations, and structural acceptance criteria approved by the NRC staff.

• RES INTERFACE INFORMATION

A 1/5 scale Mark I test program was conducted by LLL to provide confirmatory pool swell load data. The final test report was submitted to PSYB from RES on 03-10-78. Additional analyses of the test data were conducted by LLL to investigate 3-D pool swell effects and were submitted to PSYB in May 1979.

• ACRS INTERFACE INFORMATION

Meetings have been held with ACRS Subcommittee on Fluid Dynamics on 9/14/79 and 11/16/79 to discuss acceptance criteria. An additional meeting may be necessary to discuss changes to the criteria. A meeting with the full committee was held on 02-08-80.

• TECHNICAL ASSISTANCE CONTRACTS

BNL will assist in the review of the hydrodynamic load definition techniques (FIN A-3108). LLL has performed work on containment safety margins, fluid structure interaction effects, and column buckling. All of the LLL work was completed in FY 1979. The BNL work scope will be complete when the SER supplement is issued.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED

• POTENTIAL PROBLEMS

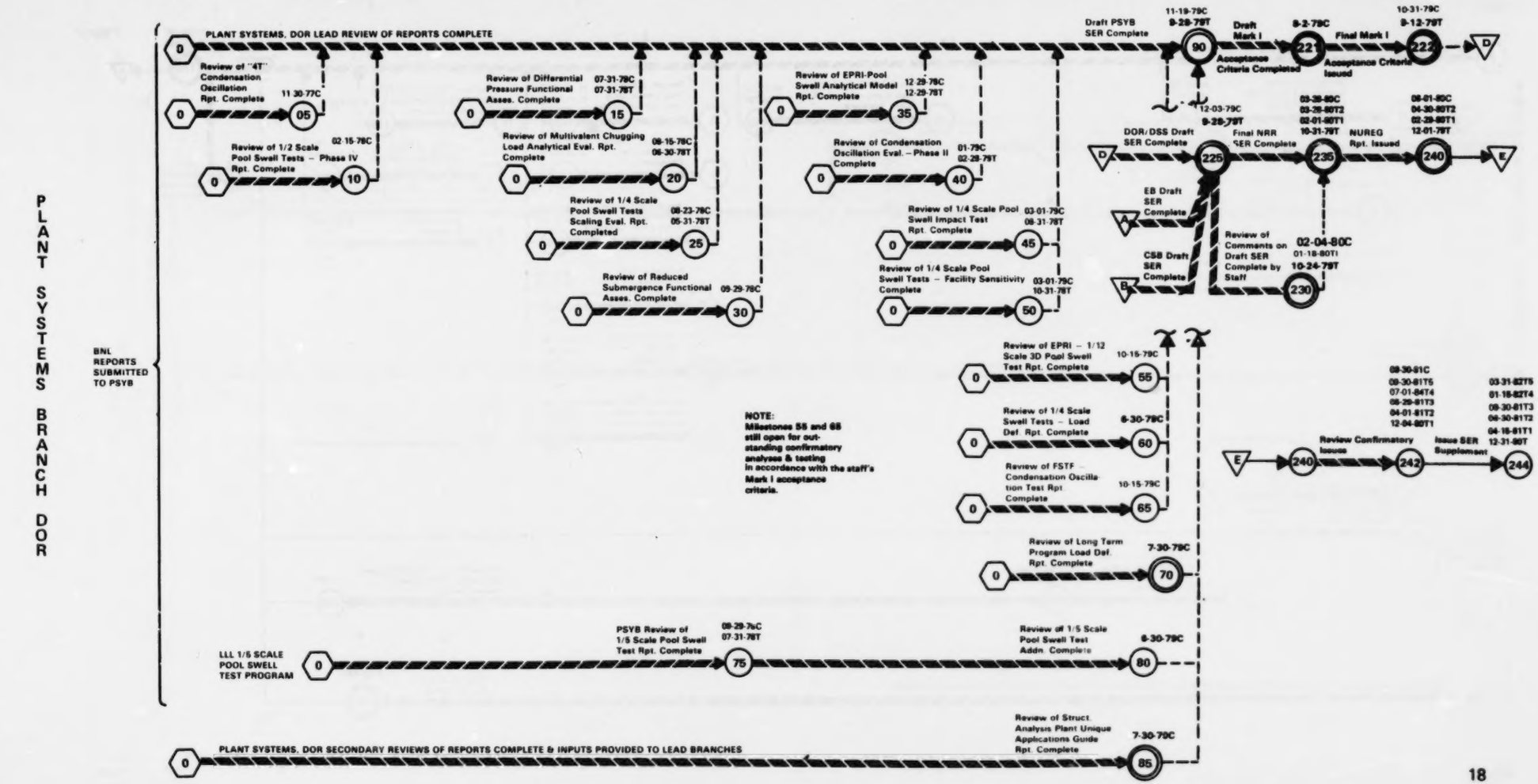
Comparisons of preliminary plant-unique analyses, using the proposed load definition techniques and the NRC acceptance criteria, with full-scale test data indicate that excessive stresses may be predicted because the loading conditions are too idealized. Confirmatory testing in the full-scale test facility (FSTF) may indicate a need for revised "condensation oscillation" loads, although none is expected.

• STATUS SUMMARY

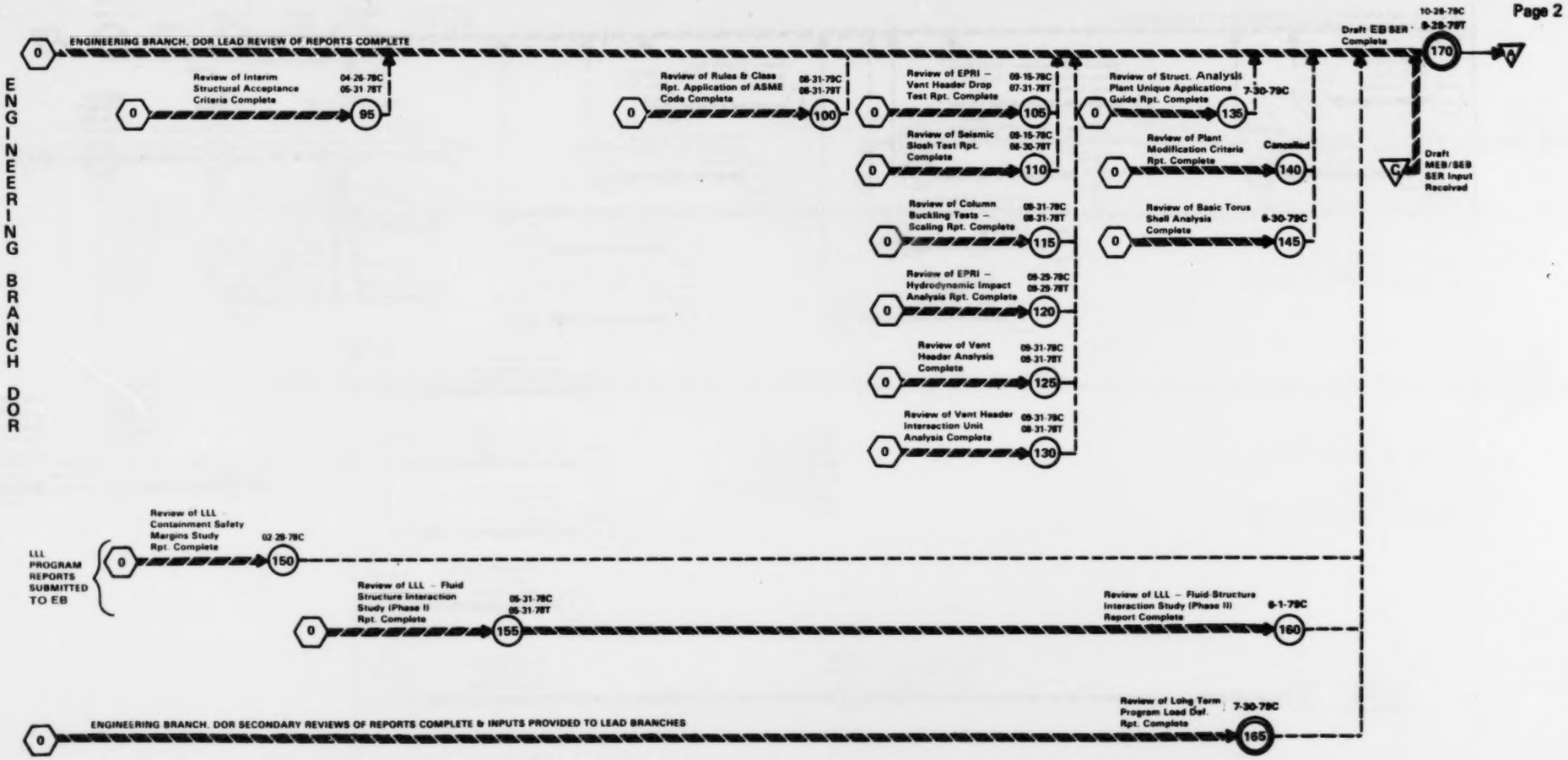
This issue is considered complete since the final NUREG was issued July, 1980, and confirmatory orders implementing the findings in NUREG-0661 were issued on January 13, 1981 for 22 operating BWRs with Mark I containments. However, BNL has reviewed three confirmatory issues for an SER supplement and issued a # draft report for NRR review. The NRR supplement is scheduled to be issued 03-31-82.

# INDICATES A CHANGE IN INFORMATION

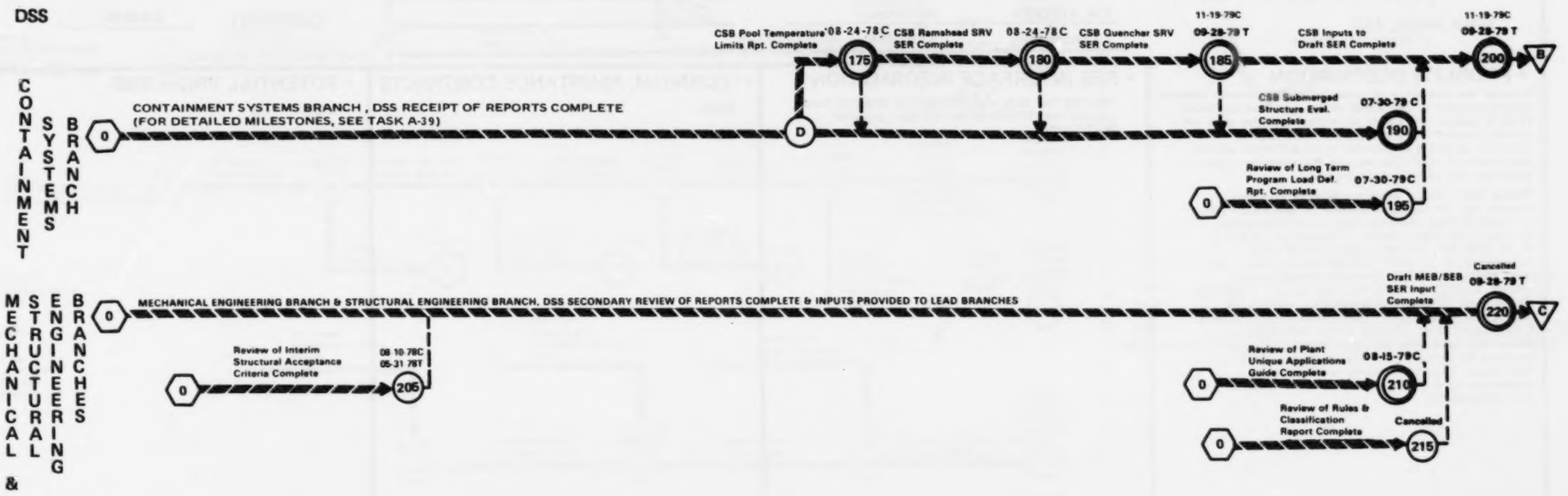
MARK I CONTAINMENT LONG-TERM PROGRAM (A-7)


PLANT SYSTEMS BRANCH DOR
BNL REPORTS SUBMITTED TO PSYB
PLANT SYSTEMS, DOR LEAD REVIEW OF REPORTS COMPLETE
Review of "4T" Condensation Oscillation Rpt. Complete 11-30-77C 05
Review of 1/2 Scale Pool Swell Tests – Phase IV Rpt. Complete 02-15-78C 10
Review of Differential Pressure Functional Asses. Complete 07-31-79C 07-31-78T 15
Review of Multivalent Chugging Load Analytical Eval. Rpt. Complete 08-15-79C 06-30-78T 20
Review of 1/4 Scale Pool Swell Tests Scaling Eval. Rpt. Completed 08-23-78C 05-31-78T 25
Review of Reduced Submergence Functional Asses. Complete 09-29-78C 30
Review of EPRI-Pool Swell Analytical Model Rpt. Complete 12-29-78C 12-29-78T 35
Review of Condensation Oscillation Eval. – Phase II Complete 01-79C 02-28-79T 40
Review of 1/4 Scale Pool Swell Impact Test Rpt. Complete 03-01-79C 08-31-78T 45
Review of 1/4 Scale Pool Swell Tests – Facility Sensitivity Complete 03-01-79C 10-31-78T 50
Draft PSYB SER Complete 11-19-79C 9-28-78T 90
Draft Mark I Acceptance Criteria Completed 8-2-78C 221
Final Mark I Acceptance Criteria Issued 10-31-79C 9-12-78T 222
12-03-79C 9-28-78T
DOR/DSS Draft SER Complete 03-28-80C 03-28-80T2 02-01-80T1 10-31-78T 225
Final NRR SER Complete 235
NUREG Rpt. Issued 08-01-80C 04-30-80T2 02-29-80T1 12-01-78T 240
EB Draft SER Complete
CSB Draft SER Complete
Review of Comments on Draft SER Complete by Staff 02-04-80C 01-18-80T1 10-24-78T 230
NOTE: Milestones 55 and 65 still open for outstanding confirmatory analyses & testing in accordance with the staff's Mark I acceptance criteria.
Review of EPRI – 1/12 Scale 3D Pool Swell Test Rpt. Complete 10-15-79C 55
Review of 1/4 Scale Swell Tests – Load Def. Rpt. Complete 6-30-79C 60
Review of FSTF Condensation Oscillation Test Rpt Complete 10-15-79C 65
Review of Long Term Program Load Def. Rpt. Complete 7-30-79C 70
LLL 1/5 SCALE POOL SWELL TEST PROGRAM
PSYB Review of 1/5 Scale Pool Swell Test Rpt. Complete 09-29-79C 07-31-78T 75
Review of 1/5 Scale Pool Swell Test Addn. Complete 6-30-79C 80
Review of Struct. Analysis Plant Unique Applications Guide Rpt. Complete 7-30-79C 85
PLANT SYSTEMS, DOR SECONDARY REVIEWS OF REPORTS COMPLETE & INPUTS PROVIDED TO LEAD BRANCHES
09-30-81C
09-30-81T5 03-31-82T5
07-01-84T4 01-18-82T4
08-29-81T3 09-30-81T3
04-01-81T2 04-30-81T2
12-04-80T1 04-16-81T1
Review Confirmatory Issues 12-31-80T
Issue SER Supplement 244
E 240 242

MARK I CONTAINMENT LONG-TERM PROGRAM (A-7)


ENGINEERING BRANCH, DOR LEAD REVIEW OF REPORTS COMPLETE
Review of Interim Structural Acceptance Criteria Complete 04-26-79C 05-31-79T 95
Review of Rules & Class Rpt. Application of ASME Code Complete 08-31-79C 08-31-79T 100
Review of EPRI – Vent Header Drop Test Rpt. Complete 09-15-79C 07-31-78T 105
Review of Seismic Slosh Test Rpt. Complete 09-15-79C 06-30-78T 110
Review of Column Buckling Tests – Scaling Rpt. Complete 08-31-79C 08-31-78T 115
Review of EPRI – Hydrodynamic Impact Analysis Rpt. Complete 09-29-79C 09-29-78T 120
Review of Vent Header Analysis Complete 08-31-79C 08-31-78T 125
Review of Vent Header Intersection Unit Analysis Complete 08-31-79C 08-31-78T 130
Review of Struct. Analysis Plant Unique Applications Guide Rpt. Complete 7-30-79C 135
Review of Plant Modification Criteria Rpt. Complete Cancelled 140
Review of Basic Torus Shell Analysis Complete 6-30-79C 145
Draft EB SER Complete 10-26-79C 9-28-78T 170
A
C
Draft MEB/SEB SER Input Received
LLL PROGRAM REPORTS SUBMITTED TO EB
Review of LLL – Containment Safety Margins Study Rpt. Complete 02-28-78C 150
Review of LLL – Fluid Structure Interaction Study (Phase I) Rpt. Complete 05-31-78C 05-31-78T 155
Review of LLL – Fluid-Structure Interaction Study (Phase II) Report Complete 6-1-79C 160
Review of Long Term Program Load Def. Rpt. Complete 7-30-79C 165
ENGINEERING BRANCH, DOR SECONDARY REVIEWS OF REPORTS COMPLETE & INPUTS PROVIDED TO LEAD BRANCHES
DOR

MARK I CONTAINMENT LONG-TERM PROGRAM (A-7)



REACTOR VESSEL MATERIALS TOUGHNESS (A-11)

AS OF WEEK ENDING FEBRUARY 19, 1982

KEY PERSONNEL

TASK MANAGER
RICHARD E. JOHNSON X24715

Richard E. Johnson

MPA ANALYST
P. GORHAM X24986

TASK REVIEWERS

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W.S. HAZELTON	MTEB/DE
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J. STROSNIDER	MEBR/RES
P.N. RANDALL	MEBR/RES
C.S. HINSON	RAB/DSI
C. SERPAN	MEBR/RES

SCHEDULED COMPLETION

1978 ANNUAL REPORT	July 1979
CURRENT	# 06-22-82

• PROBLEM DESCRIPTION

Because the possibility of failure of nuclear reactor pressure vessels designed to the ASME Boiler and Pressure Vessel Code is remote, the design of nuclear facilities does not provide protection against reactor vessel failure. However, as plants accumulate more and more service time, neutron irradiation reduces the material fracture toughness and initial safety margins.

Results from reactor vessel surveillance programs indicate that up to approximately 20 operating PWRs will have beltline materials with marginal toughness, relative to the requirements of Appendices G and H of 10 CFR Part 20, after comparatively short (approximately 10 EFPY) periods of operation. For most plants now in the licensing process, current criteria, together with the materials currently employed, are adequate to ensure suitable safety margins for reactor vessels throughout their design lives. However, a few plants under licensing review have reactor vessels that have been identified as having the potential for marginal fracture toughness within their design lives; these vessels will have to be reevaluated in the light of the new criteria for long term acceptability.

• RES INTERFACE INFORMATION

Fracture test data was provided by RES and incorporated into NUREG-0744; RES has reviewed NUREG-0744.

• ACRS INTERFACE INFORMATION

Copies of NUREG-0744 were forwarded to the ACRS when issued for public comment; no other action planned.

• TECHNICAL ASSISTANCE CONTRACTS

NONE

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
B8486	W.U.T.A.	$103,000	$ 83,088
B0737	ORNL	$415,000	$331,375
A8419	INEL	$100,000	$ 88,740
B8808	NRL	$ 95,650	$ 95,650

• POTENTIAL PROBLEMS

NONE

• STATUS SUMMARY

NUREG-0744 has been completed and was issued for public comment in September 1981.

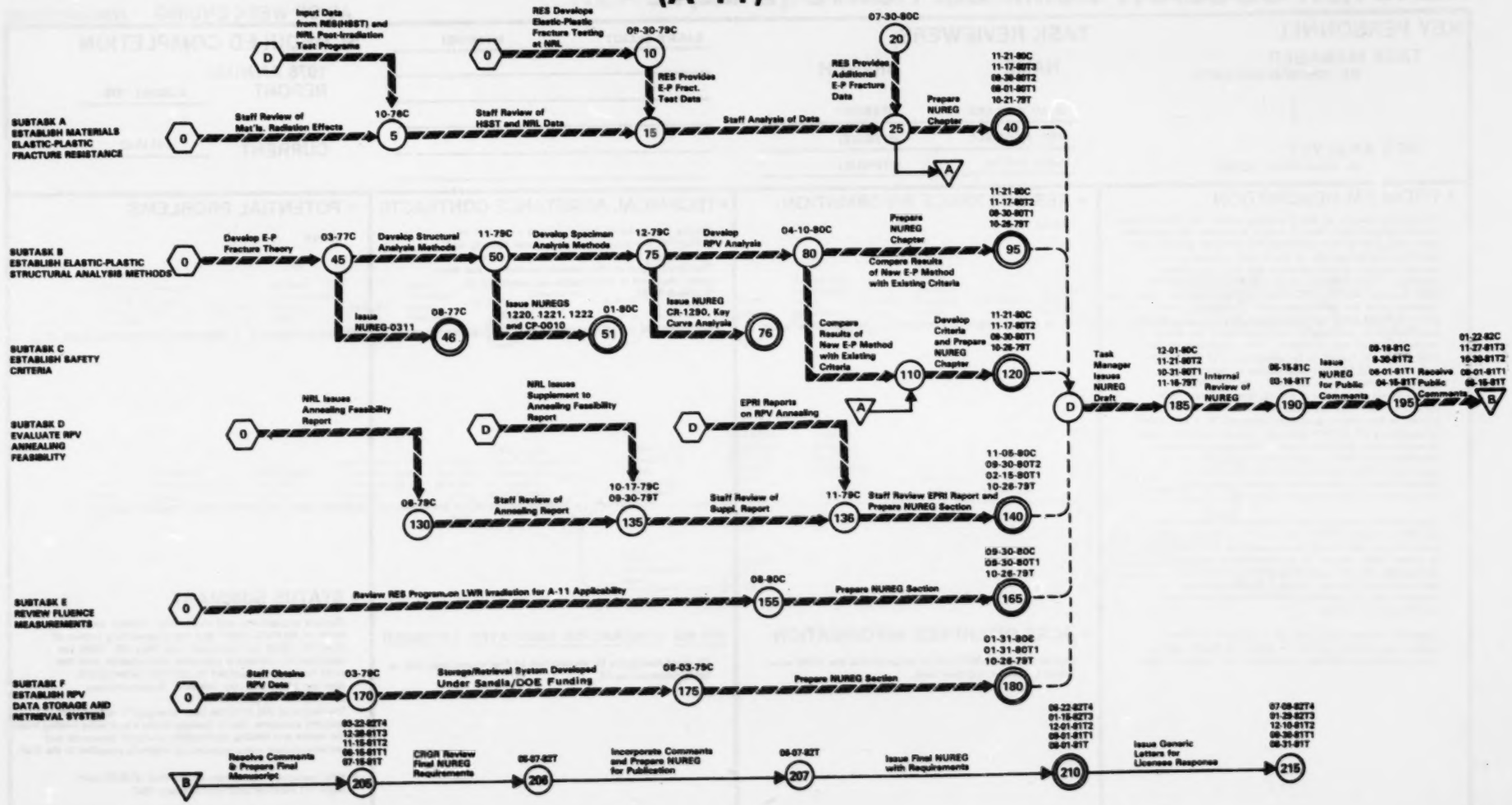
Volume 1 is a summary description of the resolution and results. Volume 2 presents a detailed description of the technical work accomplished.

Many comments were received, most of which will be resolved with editorial clarification. However, the number of comments and the complexities will require more time than previously estimated.

ASME Code Section XI Subgroup on Evaluation Standards will assist in working out the details.

CRGR review has been included after the Final NUREG has been prepared.

REACTOR VESSEL MATERIALS TOUGHNESS (A-11)



FRACTURE TOUGHNESS OF STEAM GENERATOR AND REACTOR COOLANT PUMP SUPPORTS (A-12)

AS OF WEEK ENDING FEBRUARY 19, 1982

KEY PERSONNEL

TASK MANAGER
R.E. JOHNSON (GIB) X24715

MPA ANALYST
M. ANDERSON X27021

TASK REVIEWERS

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NEIL RANDALL	MTEB(DE)
DAVE SMITH	MTEB(DE)
BARRY ELLIOT	MTEB(DE)

SCHEDULED COMPLETION

1978 ANNUAL REPORT — AUGUST 1979

CURRENT — #07-19-82

• PROBLEM DESCRIPTION

During the course of the licensing action for North Anna Power Station Unit Nos. 1 and 2 a number of questions were raised as to the potential for lamellar tearing and low fracture toughness of the steam generator and reactor coolant pump support materials for those facilities. Two different steel specifications (ASTM A36-70a and ASTM A572-70a) covered most of the material used for these supports. Toughness tests, not originally specified and not in the relevant ASTM specifications, were made on those heats for which excess material was available. The toughness of the A36 steel was found to be adequate, but the toughness of the A572 steel was relatively poor at an operating temperature of 80°F. In the case of North Anna Unit Nos. 1 and 2, the applicant has agreed to raise the temperature of the ASTM A572 beams in the steam generator supports to a minimum temperature of 225°F prior to reactor coolant system pressurization to levels above 1000 psig. Auxiliary electrical heat will be supplied as necessary to supplement the heat derived from the reactor coolant loop to obtain the required operating temperature of the support materials.

Since similar materials and designs have been used on other nuclear plants, the concerns regarding the supports for the North Anna facilities are applicable to other PWR plants. It was therefore necessary to reassess the fracture toughness of the steam generator and reactor coolant pump support materials for all operating PWR plants and those in CP and OL review.

Lamellar tearing may also be a problem in those support structures similar in design to North Anna. This possibility will be investigated on a generic basis.

• RES INTERFACE INFORMATION

NONE

• ACRS INTERFACE INFORMATION

Copies of NUREG-0577 will be forwarded to the ACRS when issued in revised and final form.

• TECHNICAL ASSISTANCE CONTRACTS

Sandia Laboratories has provided expertise in evaluating the potential for lamellar tearing and low fracture toughness of the support materials. The final input to the generic resolution has been submitted and is included as an appendix to NUREG-0577.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED

Contract assistance for preparation of final report from FIRL is being managed by DL.

• POTENTIAL PROBLEMS

NONE

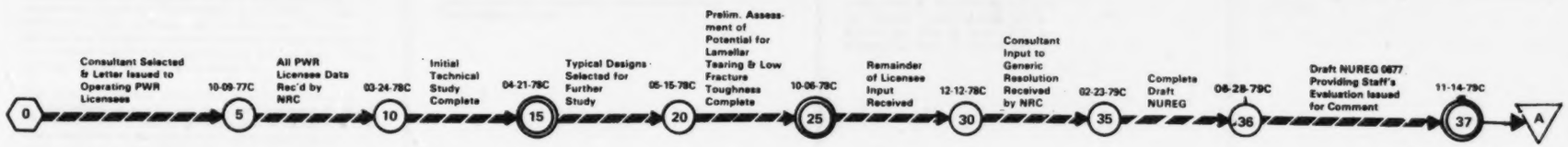
• STATUS SUMMARY

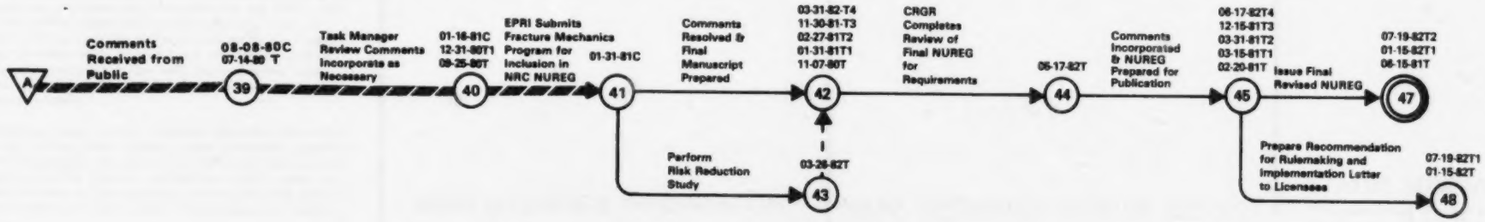
Review procedures and evaluation criteria were set forth in NUREG-0577 and the augmenting letters of May 19, 1980 (to licensees), and May 20, 1980 (to applicants). Industry response was adverse, and the staff has been requested to consider alternatives, first by EPRI and by the AIF Subcommittee on Material Requirements.
The scope of USI A-12 has been changed to reflect staff and industry concerns. Major changes include removing bolting from the scope and limiting applicability to steam generator and primary coolant pump supports as originally specified in the TAP.

NRC reviewers' comments on the final NUREG draft were not received until mid-January, 1982.

Allowance for CRGR review has been incorporated into the schedule.

FRACTURE TOUGHNESS OF STEAM GENERATOR AND REACTOR COOLANT PUMP SUPPORTS (A-12)





SYSTEMS INTERACTIONS IN NUCLEAR POWER PLANTS (A-17)

AS OF WEEK ENDING FEBRUARY 19, 1982

KEY PERSONNEL

TASK MANAGER
PAUL NORIAN (Acting) X27487

Paul Norian

MPA ANALYST
M. ANDERSON X27021

TASK REVIEWERS

NAME	BRANCH
NO ASSIGNED REVIEWERS	

SCHEDULED COMPLETION

1978 ANNUAL REPORT Phase I - 09-79

CURRENT N/A

• PROBLEM DESCRIPTION

The design of a nuclear power plant is accomplished by groups of engineers and scientists organized into engineering disciplines and into scientific disciplines. The reviews performed by the designers include interdisciplinary reviews to assure the functional compatibility of the plant structures, systems, and components. Safety reviews and accident analyses provide further assurance that system functional requirements will be met. These reviews include failure mode analyses.

The NRC review and evaluation of safety systems is accomplished in accordance with the Standard Review Plan (SRP) which assigns primary and secondary review responsibilities to organizational units arranged by plant systems or by disciplines. Each element of the SRP is assigned to an organizational unit for primary responsibility and, where appropriate, to other units for secondary responsibilities.

Thus, the design and analyses by the plant designers, and the subsequent review and evaluation by the NRC staff take into consideration the interdisciplinary areas of concern and account for systems interaction to a large extent. Furthermore, many of our regulatory criteria are aimed at controlling the risks from systems interactions. Examples include the single failure criterion and separation criteria.

Nevertheless, there is some question regarding the interaction of various plant systems, both as to the supporting roles such systems play and as to the effect one system can have on other systems, particularly with regard to whether actions or consequences could adversely affect the presumed redundancy and independence of safety systems.

The problem to be resolved by this task is to identify where the present design, analysis, and review procedures may not acceptably account for potentially adverse systems interaction and to recommend the regulatory action that should be taken to rectify deficiencies in the procedures.

• RES INTERFACE INFORMATION

The Division of Risk Analysis has been consulted during the development and execution of this plan. The Division of Engineering Technology will participate in the development of a Regulatory Guide for Systems Interaction Analyses.

• ACRS INTERFACE INFORMATION

The review is under the cognizance of the ACRS Subcommittee on Plant Arrangements. A meeting was held 02-20-80 in Washington, D.C. with the Subcommittee which recommended a demonstration of Phase I methodology before proceeding much further.

A meeting is scheduled for 02-26-82 with the ACRS Subcommittee on Indian Point-3 to describe the PASNY systems interaction program on Indian Point-3.

• TECHNICAL ASSISTANCE CONTRACTS

Phase I work was performed by Sandia National Laboratories under FIN # A-1113(9) by NRR for $219K and under FIN # A-1084(8) by OSD for $320K. Expenditures were $197K for FIN # A-1113(9) and $320K for FIN # A-1084(8).

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
# A0403	LLNL	$150,000	$ 29,000
A3365	BNL	$ 50,000	
# B2329	PNL	$150,000	
A1295	SANDIA	$ 27,000	
A-495	LLNL	$183,000	$129,000
A-3358	BNC	$ 80,000	$ 80,000
B-2330	PNL	$ 80,000	$ 80,000

• POTENTIAL PROBLEMS

PASNY may delay initiation of Indian Point-3 study.

LWRs for implementation of interim guidance have not yet been selected.

• STATUS SUMMARY

The anticipated Phase II program will no be pursued as an USI. Work originally planned under TAP A-17 will now be performed under TMI Action Plan Item II-C-3, Systems Interaction. This activity will now be managed by the Systems Interaction Section, Reliability and Risk Assessment Branch, DST. The Section Leader is F. Coffman, X27487 The status of this item is reported in the quarterly updates of the TMI Action Plan Tracking System (APTS).

PASNY's final program description for Indian Point-3 was received 01-20-82. Interim guidance for implementation on selected plants is scheduled for completion in 07/82.

# INDICATES A CHANGE IN INFORMATION

DETERMINATION OF SAFETY RELIEF VALVE (SRV) POOL DYNAMIC LOADS AND TEMPERATURE LIMITS FOR BWR CONTAINMENT (A-39)

AS OF WEEK ENDING FEBRUARY 19, 1982

KEY PERSONNEL

TASK MANAGER
T.M. SU (GIB) X27477

MPA ANALYST
M. ANDERSON X27021

TASK REVIEWERS

NAME	BRANCH

SCHEDULED COMPLETION

1978 ANNUAL REPORT — October 1979

CURRENT
MARK I - 02-29-80C
MARK II - 06-15-82
MARK III - 06-15-82

• PROBLEM DESCRIPTION

Operation of BWR primary system pressure relief valves can result in hydrodynamic loads on the suppression pool retaining structures or structures located within the pool. These loads result from:

1. Initial vent clearing of a relief valve pipe; and
2. Steam quenching due to locally high pool temperature.

• RES INTERFACE INFORMATION

NONE

• ACRS INTERFACE INFORMATION

Discussions with the ACRS Subcommittee on Mark III SRV loads and Kuoshong inplant test results was presented on September 25, 1981.

• TECHNICAL ASSISTANCE CONTRACTS

BNL has completed Mark II T-Quencher and Mark III X-Quencher review and is preparing input to the NUREG report.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
A3098	BNL	FY 79 $85,000 FY 80 $50,000	
A3321	BNL	FY 80 $35,000 FY 81 $70,000	

• POTENTIAL PROBLEMS

NONE

• STATUS SUMMARY

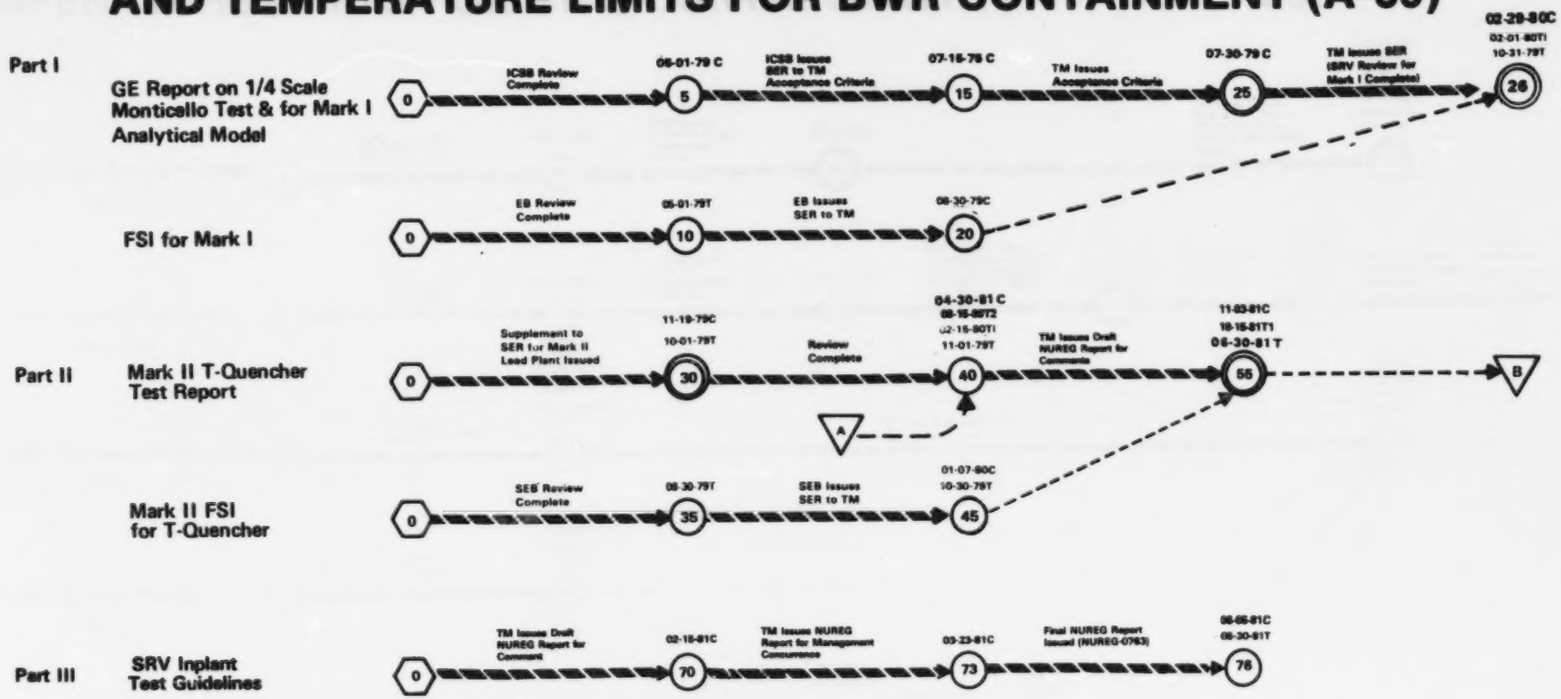
Task A-39 Mark I effort completed in February 1980.

NUREG report for inplant SRV test was issued in June 1981.

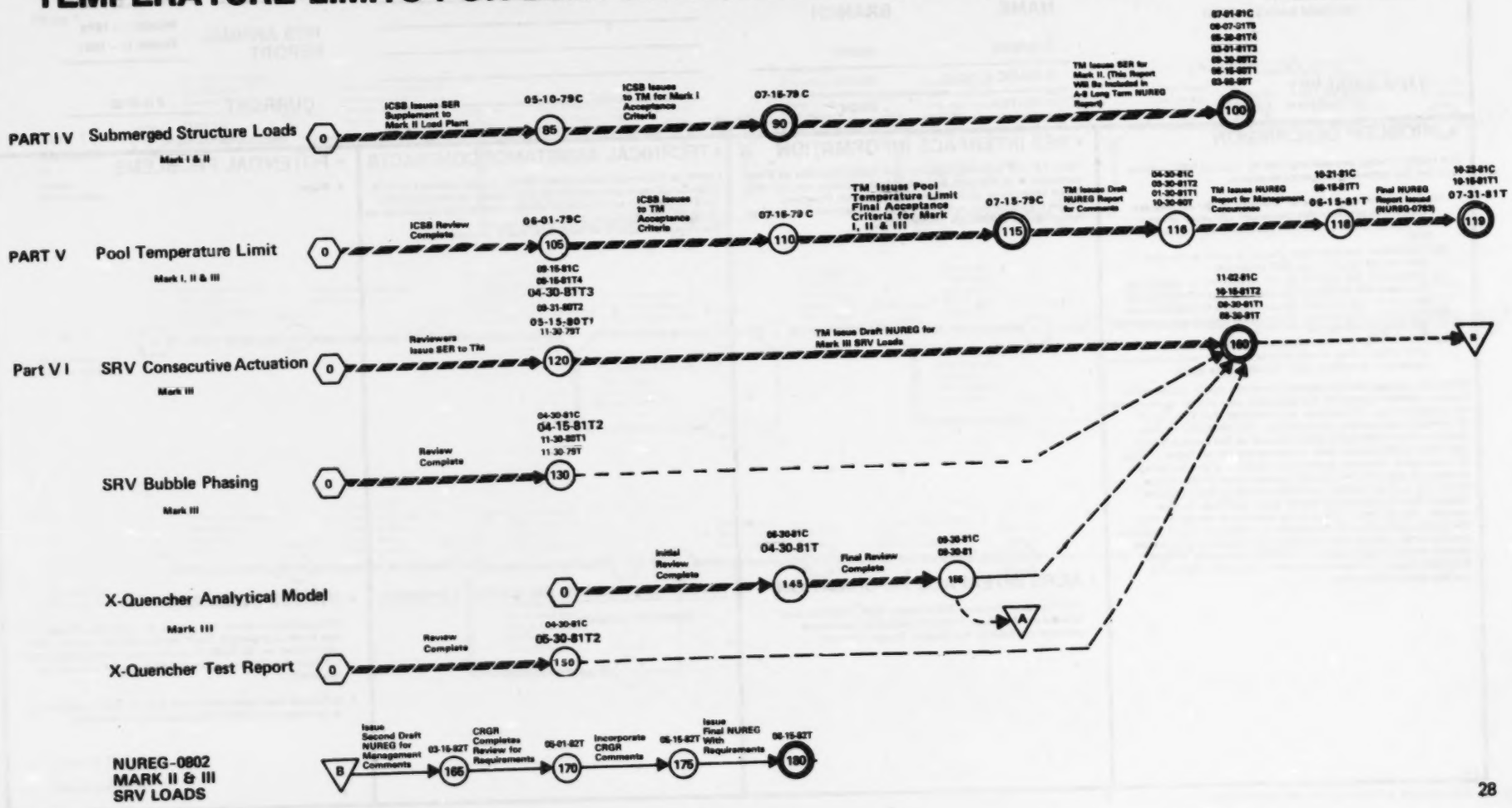
NUREG report for suppression pool temperature limit was approved by EDO in 10/81. Issuance of the report is pending OMB approval.

Draft NUREG report for Mark II and III SRV loads was issued for comments and management concurrence.

DETERMINATION OF SAFETY RELIEF VALVE (SRV) POOL DYNAMIC LOADS AND TEMPERATURE LIMITS FOR BWR CONTAINMENT (A-39)



DETERMINATION OF SAFETY RELIEF VALVE (SRV) POOL DYNAMICS LOADS AND TEMPERATURE LIMITS FOR BWR CONTAINMENT (A-39) Continued



*Includes recommended changes to licensing guidance documents

SEISMIC DESIGN CRITERIA - SHORT TERM PROGRAM (A-40)

AS OF WEEK ENDING FEBRUARY 19, 1982

KEY PERSONNEL

TASK MANAGER
GOUTAM BAGCHI X28251

Goutam Bagchi

MPA ANALYST
P. GORHAM X24986

TASK REVIEWERS

NAME	BRANCH
T. CHENG	SEB/DL
S. CHAN, H. POLK	SEB/DE
L. REITER	GB/DE

SCHEDULED COMPLETION

1978 ANNUAL REPORT	PHASE I – 1979 PHASE II – 1981
CURRENT	# 12-31-82

• PROBLEM DESCRIPTION

The seismic design process required by current NRC criteria includes the following sequence of events:

(a) Define the magnitude or intensity of the earthquake which will produce the maximum vibratory ground motion at the site (the safe shutdown earthquake or SSE).
(b) Determine the free-field ground motion at the site that would result if the SSE occurred.
(c) Determine the motion of site structures by modifying the free-field motion to account for the interaction of the site structures with the underlying foundation soil.
(d) Determine the motion of the plant equipment supported by the site structures.
(e) Compare the seismic loads, in appropriate combination with other loads, on structures, systems, and components important to safety, with the allowable loads.

While this seismic design sequence includes many conservative factors, certain aspects of the sequence may not be conservative for all plant sites. At present, it is believed that the overall sequence is adequately conservative. The objective of this program is to investigate selected areas of the seismic design sequence to determine their conservatism for all types of sites, to investigate alternate approaches to parts of the design sequence, to quantify the overall conservatism of the design sequence, and to modify the NRC criteria in the Standard Review Plan if changes are found to be justified. In this manner, this program will provide additional assurance that the health and safety of the public is protected, and if possible, reduce costly design conservatisms by improving (1) current seismic design requirements, (2) NRR's capability to quantitatively assess the overall adequacy of seismic design for nuclear plants in general.

• RES INTERFACE INFORMATION

Task 1.0 will provide input to the Quantification of Conservatisms in Present Seismic Methodology, one of the three objectives of the Seismic Research Program. Close coordination with RES will be maintained through meetings to exchange information.

• ACRS INTERFACE INFORMATION

The Extreme External Phenomena Subcommittee was briefed on the Phase I reports including the recommended changes to the Seismic Design Criteria.

• TECHNICAL ASSISTANCE CONTRACTS

Lawrence Livermore Laboratory (LLL), under contract to RES, reviewed all reports by 04-30-79. LLL report on recommendations for changes to the seismic design criteria was completed on 12-28-79. (NUREG/CR-1161).
The contractor reports on tasks 7, 8, and 9 are complete.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED

Contracts complete.

• POTENTIAL PROBLEMS

None.

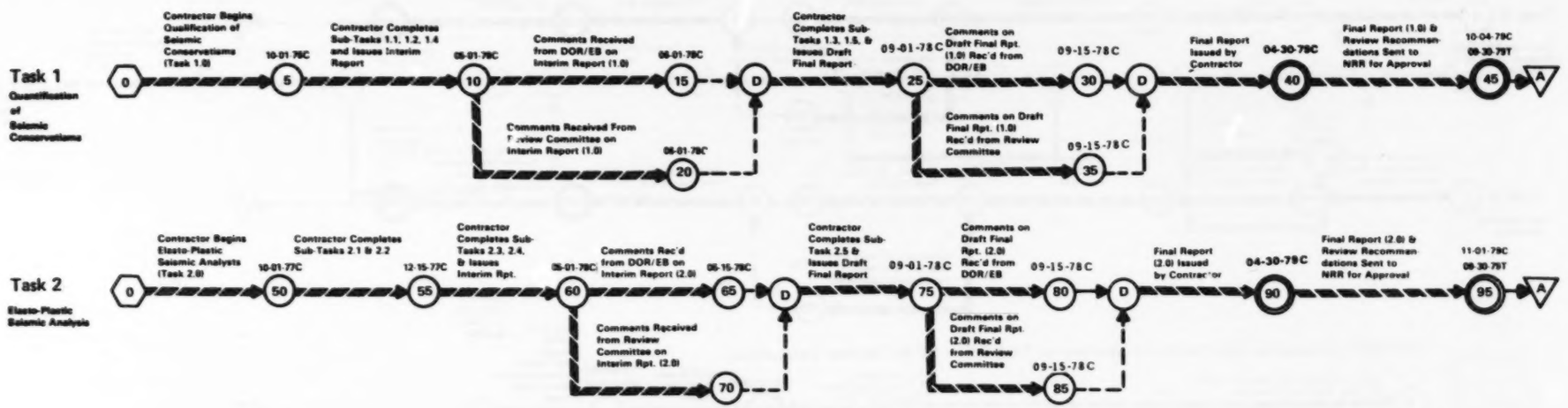
• STATUS SUMMARY

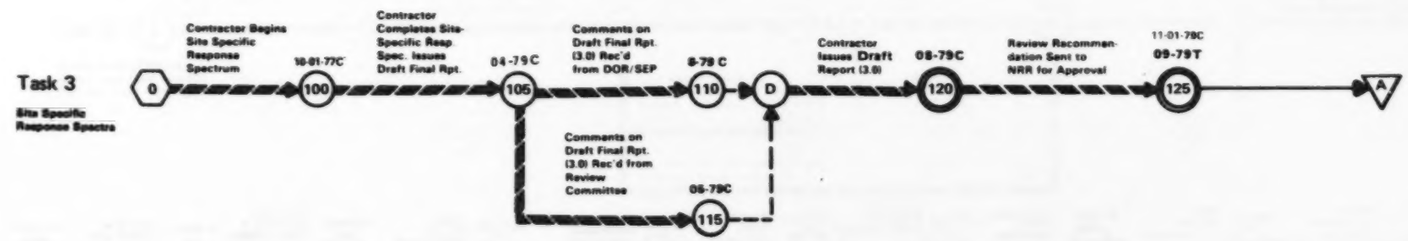
All contractor work necessary to complete Phase I has been completed. Staff review of LLL report was completed on 10-30-80. A draft report presenting staff position has been prepared and is under review by the staff.

Schedule has been extended due to new requirement to incorporate CRGR implementation reviews.

SEISMIC DESIGN CRITERIA - SHORT TERM PROGRAM (A-40)

Phase 1

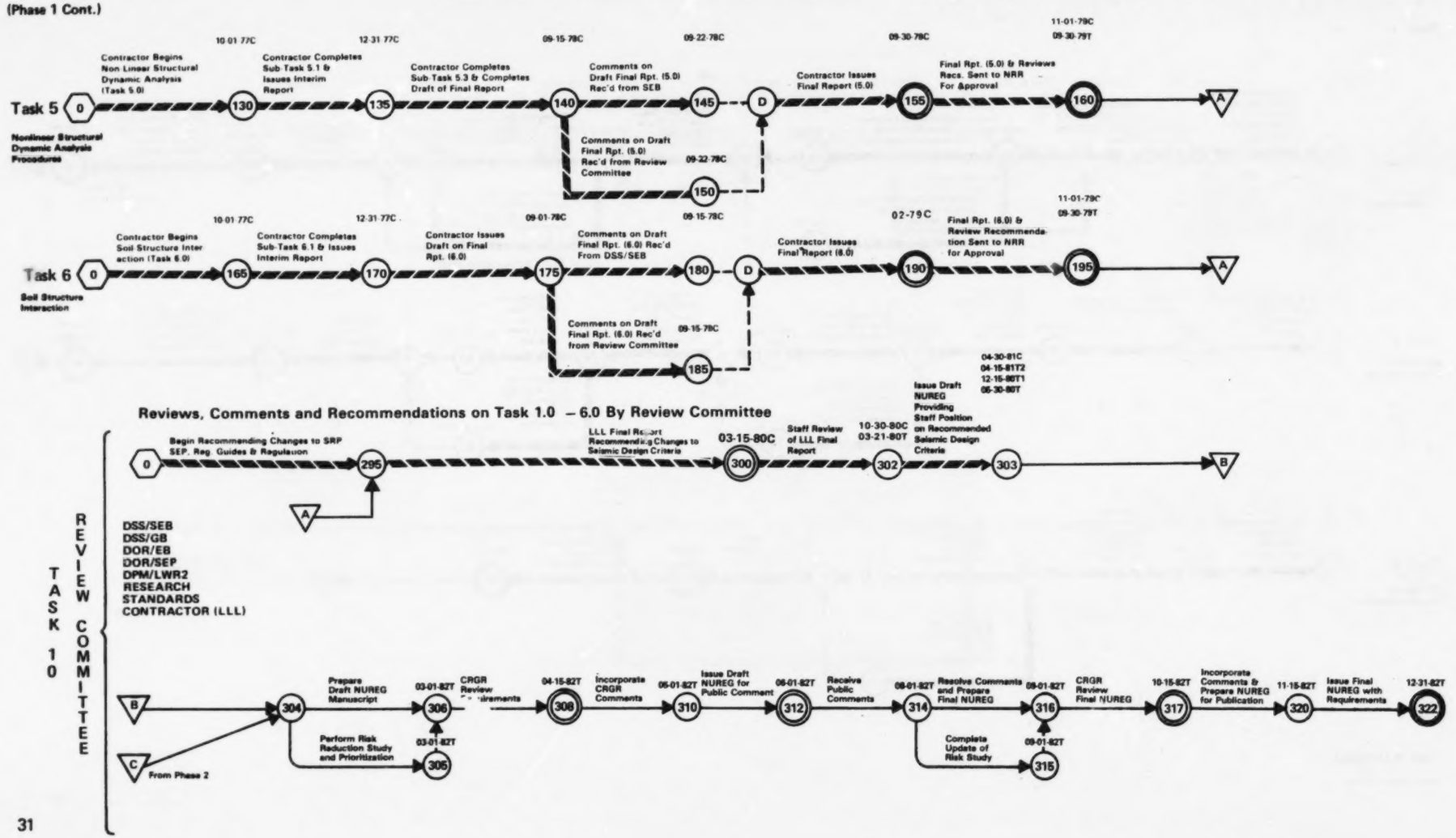




Task 4 Deleted

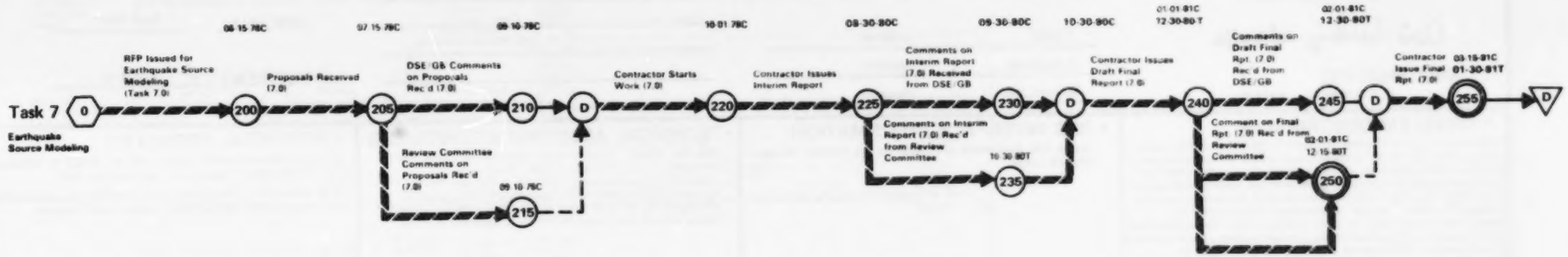
Seismic After Shocks

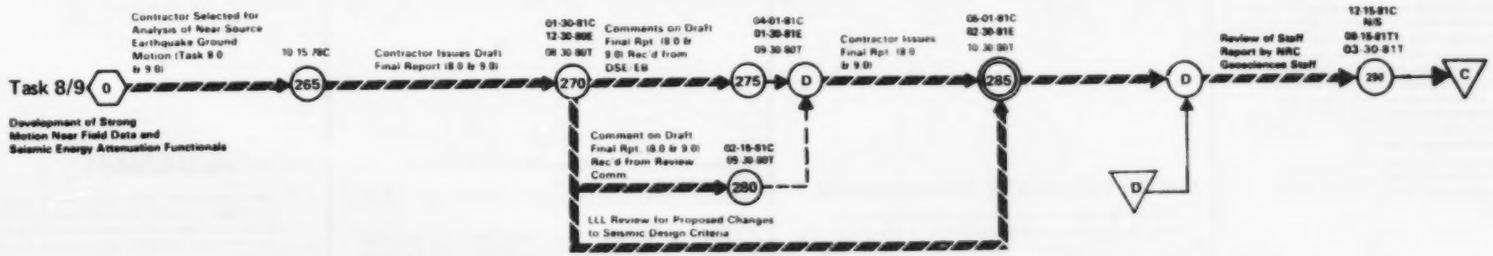
SEISMIC DESIGN CRITERIA - SHORT TERM PROGRAM (A-40)



SEISMIC DESIGN CRITERIA - SHORT TERM PROGRAM (A-40)

Phase 2





CONTAINMENT EMERGENCY SUMP PERFORMANCE (A-43)

AS OF WEEK ENDING FEBRUARY 19, 1982

KEY PERSONNEL

TASK MANAGER
A. SERKIZ (DST) X24217

A W Serkiz 2/26/82

MPA ANALYST
D. SCHWARZ X27106

TASK REVIEWERS

NAME	BRANCH
F. ORR	RSB/DSI
P. NORIAN	GIB/DST
W. BUTLER	CSB/DSI

SCHEDULED COMPLETION

ORIGINAL: April 1982

CURRENT: # 04-01-83

• PROBLEM DESCRIPTION

Following a Loss of Coolant Accident (LOCA) in a PWR, water flowing from the break in the primary system would collect on the floor of containment. During the injection mode, water for core cooling and containment spray is drawn from a large supply tank. When the water reached a low level in the tank, pumps are realigned to draw from the containment. This is called the recirculation mode wherein water is drawn from the containment floor or sump and pumped to the primary system or containment spray headers. This program addresses the safety issue of adequate sump or suppression pool function in the recirculation mode. It is the objective of this program to develop improved criteria for design, testing, and evaluation which will provide better assurance that emergency sumps will function to satisfy system requirements.

The principal concerns are somewhat interrelated but are best discussed separately. One deals with the various kinds of insulation used on piping and components inside of containment. The concern being that break-initiated debris from the insulation could cause blockage of the sump or otherwise adversely affect the operation of the pumps, spray nozzles, and valves of the safety systems.

The second deals with the hydraulic performance of the sump as related to the hydraulic performance to safety systems supplied therefrom. Preoperational tests have been performed on a number of plants to demonstrate operability in the recirculation mode. Adverse flow conditions have been encountered requiring design and procedural modifications to eliminate them. These conditions, air entrainment, cavitation, and vortex formation, are aggravated by blockage. If not avoided or suppressed, they could result in pump failure during the long term cooling phase following a LOCA.

The concerns relative to debris, blockage, and hydraulic performance also apply to boiling water reactors during recirculation from the suppression pools, and will also be addressed.

• RES INTERFACE INFORMATION

None; USI A-43 being managed by the Generic Issues Branch (GIB).

• ACRS INTERFACE INFORMATION

The ACRS staff is kept abreast of findings and developments on an informal basis. A briefing on the # current findings and status of resolution was held December 3, 1981.

• TECHNICAL ASSISTANCE CONTRACTS

FIN No. A1237, "Containment Emergency Sump Performance", and FIN No. A1296, "Technical Assistance for Resolution of USI A-43", are being funded by RES and NRR respectively. This work is managed by the GIB Task Manager and these combined efforts are expected to be concluded in FY82.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
FY 81:			
# A1237	Sandia	$225,000	$101,000
# A1296	Sandia	$120,000	$ 75,100
FY 82:			
# A1237	Sandia	$209,000*	$ 85,000 thru 1/82
# A1296	Sandia	$150,000	$ 73,400 thru 1/82

* This 1/12/82 RES obligation of $209,000 is less than the $275,000 guidance set forth to Sandia and insufficient to conclude FY 82 efforts related to A1237.

• POTENTIAL PROBLEMS

1) Insufficient funding under FIN No. A1237 to conclude FY 82 efforts leading to resolution of A-43.

2) Slippage of final resolution date due to uncertainties associated with CRGR reviews and releases.

• STATUS SUMMARY

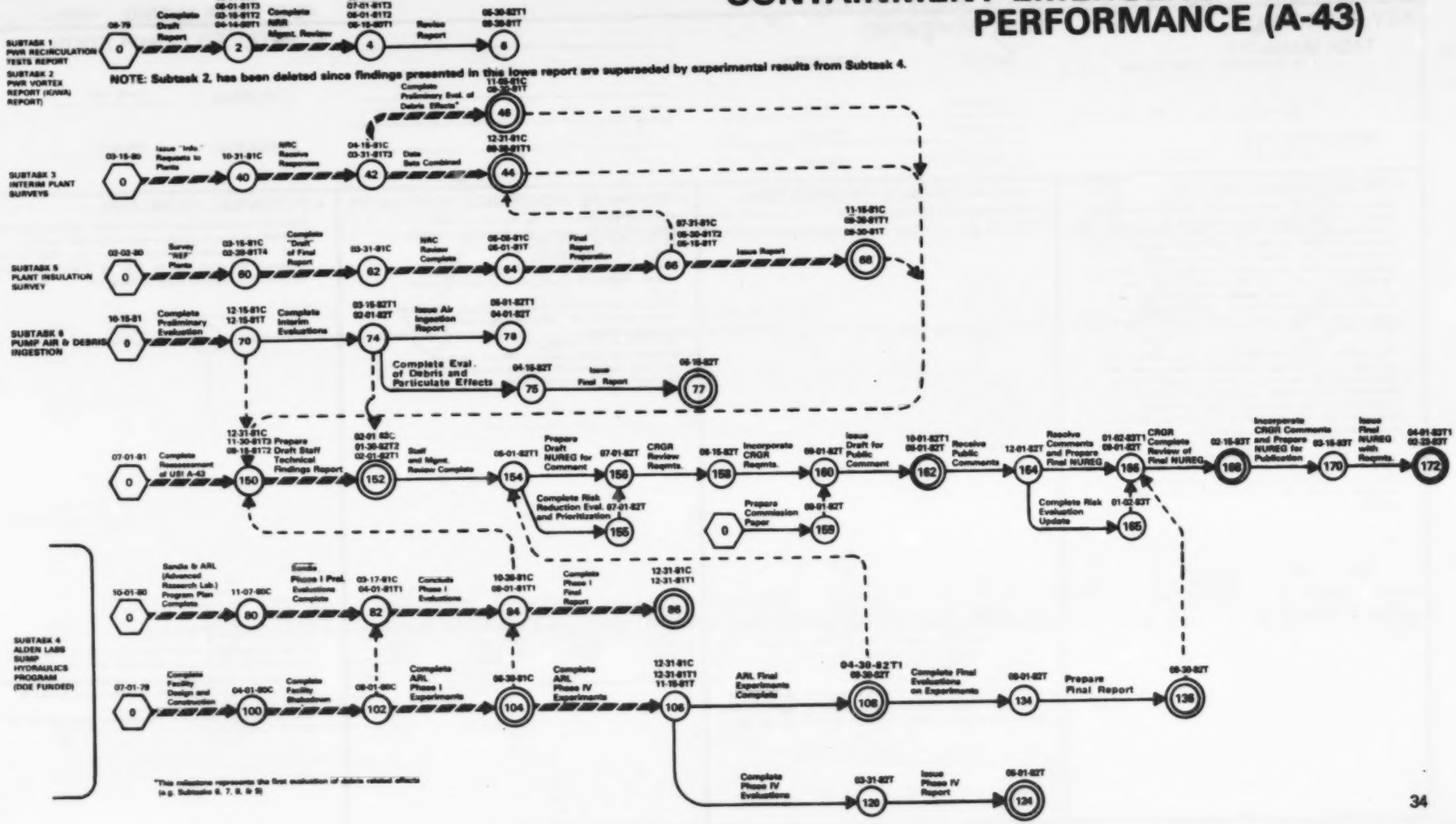
1) Alden Research Laboratory (ARL) tests nearing completion. April 1982 test completion projected. Results continue to show low levels of air ingestion and therefore a much less severe hydraulic situation than previously hypothesized.

2) Plant insulation usage surveys have been completed. Results are being reviewed internally. Debris estimates have been shown to be plant specific. Discussions with Maine Yankee (1 of 5 plants analyzed) are continuing.

3) Pump air and particulate ingestion effects are being evaluated. Results to date show that for levels of 2-4% air void, pumping capacity is not significantly degraded. Discussions with pump manufacturers are planned to obtain industry experience data.

# INDICATES A CHANGE IN INFORMATION

CONTAINMENT EMERGENCY SUMP PERFORMANCE (A-43)


SUBTASK 1 PWR RECIRCULATION TESTS REPORT
SUBTASK 2 PWR VORTEX REPORT (IDAHO REPORT)
NOTE: Subtask 2, has been deleted since findings presented in this lowa report are superseded by experimental results from Subtask 4.
SUBTASK 3 INTERIM PLANT SURVEYS
SUBTASK 5 PLANT INSULATION SURVEY
SUBTASK 6 PUMP AIR & DEBRIS INGESTION
Complete Preliminary Evaluation
Complete Interim Evaluations
Issue Air Ingestion Report
Complete Eval. of Debris and Particulate Effects
Issue Final Report
SUBTASK 4 ALDEN LABS SUMP HYDRAULICS PROGRAM (DOE FUNDED)
Complete Facility Design and Construction
Complete Facility Subdivision
Complete ARL Phase I Experiments
Complete ARL Phase IV Experiments
ARL Final Experiments Complete
Complete Final Evaluations on Experiments
Prepare Final Report
Complete Phase IV Evaluations
Issue Phase IV Report
Complete Draft Report
Complete NRR Mgmt. Review
Revise Report
Complete Preliminary Eval. of Debris Effects*
Issue "Info Requests to Plants
NRC Receive Responses
Data Sets Combined
Survey "REF" Plants
Complete "Draft" of Final Report
NRC Review Complete
Final Report Preparation
Issue Report
Complete Assessment of USI A-43
Prepare Draft Staff Technical Findings Report
Staff and Mgmt. Review Complete
Prepare Draft NUREG for Comment
CRGR Review Reqmts.
Incorporate CRGR Reqmts.
Issue Draft for Public Comment
Receive Public Comments
Resolve Comments and Prepare Final NUREG
CRGR Complete Review of Final NUREG
Incorporate CRGR Comments and Prepare NUREG for Publication
Issue Final NUREG with Reqmts.
Complete Risk Reduction Eval. and Prioritization
Complete Risk Evaluation Update
Prepare Commission Paper
Sandia Phase I Prel. Evaluations Complete
Conclude Phase I Evaluations
Complete Phase I Final Report
*This milestone represents the first evaluation of debris related effects (e.g. Subtasks 6, 7, 8, or 9)

STATION BLACKOUT (A-44)

AS OF WEEK ENDING FEBRUARY 19, 1982

KEY PERSONNEL

TASK MANAGER
P. BARANOWSKY (RES) X35960

P. Baranowsky for P. Baranowsky

MPA ANALYST
D. SCHWARZ X27106

TASK REVIEWERS

NAME	BRANCH
C. NELSON	ORB/DL
# O. CHOPRA	PSB/DSI
F. CLEMENSON	ASB/DSI

SCHEDULED COMPLETION

ORIGINAL JUNE 1982

CURRENT # 06-30-83

• PROBLEM DESCRIPTION

Electric power for safety systems at nuclear power plants is supplied by two redundant and independent divisions. Each of these electrical divisions includes an offsite alternating current (A.C.) source, an onsite A.C. source (usually diesel-generators), and a direct current (D.C.) source. Appendix A to 10 CFR 50 defines a total loss of offsite power as an anticipated occurrence, and as such, it is required that an independent emergency onsite power supply be provided at nuclear power plants.

The unlikely, but possible loss of A.C. power (that is, the loss of A.C. power from the offsite source and from the onsite source) is referred to as a station blackout. In the event of a station blackout, the capability to cool the reactor core would be dependent on the availability of systems which do not require A.C. power supplies, and on the ability to restore A.C. power in a timely manner. The concern is that the occurrence of a station blackout may be a relatively high probability event and that the consequences of this event may be unacceptable, for example, severe core damage may result.

• RES INTERFACE INFORMATION

RES is managing the Task for NRR

• ACRS INTERFACE INFORMATION

Station Blackout is related to a number of ACRS concerns regarding the reliability of power systems. This task will be coordinated with the committee as the task progresses.

Task action plan outline was discussed with AC/DC Power Reliability Subcommittee on January 22, 1981. Another subcommittee briefing was held on October 30, 1981.

• TECHNICAL ASSISTANCE CONTRACTS

Technical Assistance in the amount of 800K is estimated for this task. The following elements are included in this program:

ORNL FIN B0744 $550K –
Evaluate expected frequency and duration of offsite (preferred) power losses at nuclear power plants.
Estimate the reliability and evaluate the dominant factors affecting the reliability of emergency A.C. power supplies.

SNL FIN A1302 $250K –
Evaluate the risks posed by station blackout accidents and assess the effectiveness of safety improvements in reducing those risks.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED*
# B0744	ORNL	$550K	$396K
# A1302	SNL	$250K	$199K

*As of December 31, 1981.

• POTENTIAL PROBLEMS

Delay in receiving approvals to obtain diesel generator reliability data has compressed scheduled work at ORNL. This resulted in slippage from Feb. 1982 for completion of the draft reports. The use of additional subcontractor personnel may be required.

• STATUS SUMMARY

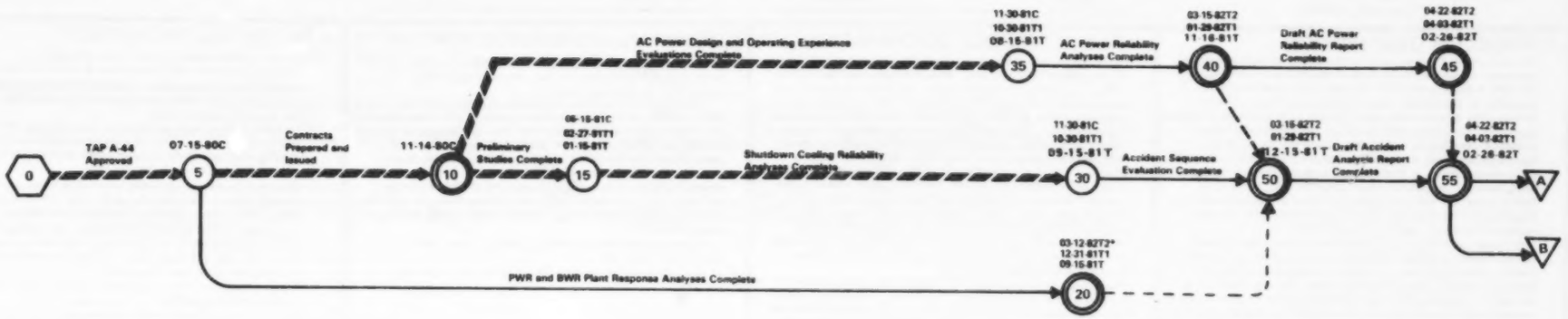
A Task Action Plan for this task was issued in July 1980 and work is proceeding in accordance with this plan.

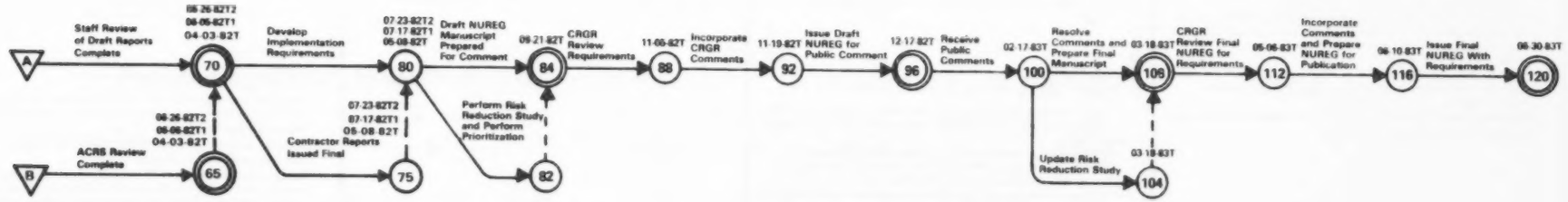
A request for diesel generator reliability data was sent to all licensees in July 1981. The first data began to arrive in October 1981.

Draft contractor technical reports are expected by the # end of April 1982.

# INDICATES A CHANGE IN INFORMATION

STATION BLACKOUT (A-44)





SHUTDOWN DECAY HEAT REMOVAL REQUIREMENTS (A-45)

AS OF WEEK ENDING FEBRUARY 19, 1982

KEY PERSONNEL

TASK MANAGER
ANDREW MARCHESE X24712

MPA ANALYST
P. GORHAM X24886

TASK REVIEWERS

NAME	BRANCH
R. LOBEL	ASB/DSI
C. GRAVES	RSB/DSI
F. ROSA	ICSB/DSI
M. SRINIVASAN	PSB/DSI
R. ANAND	CEB/DE
E. McPEEK	SSPB/DL
C. NELSON	ORB-3/DL
S. ISRAEL	RRAB/DST
M. TAYLOR	DRA/RES

SCHEDULED COMPLETION

	10/85 PRELIMINARY NETWORK
ORIGINAL	
CURRENT	# N/S

• PROBLEM DESCRIPTION

Although many improvements to the steam generator auxiliary feedwater system were required of the reactor manufacturers by the NRC following the TMI-2 accident, the staff feels that providing an alternative means of decay heat removal could substantially increase the plants' capability to deal with a broader spectrum of transients and accidents and potentially could, therefore, significantly reduce the overall risk to the public. Consequently, Task A-45 will investigate alternative means of decay heat removal in PWR plants, including but not limited to using existing equipment where possible. This Unresolved Safety Issue will also investigate the need and possible design requirements for improving reliability of decay heat removal systems in boiling water reactors (BWRs).

The overall purpose of Task A-45 is to evaluate the adequacy of current licensing design requirements, in order to ensure that nuclear power plants do not pose an unacceptable risk due to failure to remove shut-down decay heat. The objective will be to develop a comprehensive and consistent set of shutdown cooling requirements for existing and future LWRs, including the study of alternative means of shutdown decay heat removal and of diverse "dedicated" systems for this purpose.

The approach taken to this problem comprises the following subtasks:

- Development of interim criteria to judge acceptability of decay heat removal systems in existing & future plants.
- Development of means for improvement of DHR systems.
- Assessment of existing plants to identify those in which DHR systems require improvement.
- Development of proposals for improvement of DHR systems in selected plants to establish feasibility and cost effectiveness.
- Reconsideration of the adequacy of the interim acceptance criteria for DHR systems.
- Development of a plan for implementing new requirements, if any, for DHR systems required to meet the acceptance criteria above.

The interrelation and relative timing of each of the sub-tasks is shown on the schedule network.

• RES INTERFACE INFORMATION

Close coordination and cooperation will be required on Task A-45 between NRR and RES. RES assistance will be required from the Divisions of Risk Analysis, Accident Evaluation, and Engineering Technology. The Division of Risk Analysis will provide technical input from their Sandia Laboratory Program on Alternate Decay Heat Removal Concepts, technical evaluations relative to reliability and risk assessment for shutdown decay heat removal systems, and input from Task A-44, "Station Blackout," relative to shutdown cooling systems. The Division of Accident Evaluation will provide technical input relative to the transient response of existing and improved shutdown decay heat removal systems to transient events and small LOCAs. This will also include performing (in-house, contractors) detailed thermal-hydraulics analyses where required to support existing and improved decay heat removal systems behavior under transient and accident conditions.

The Division of Engineering Technology will provide assistance in the preparation and publication (i.e., Rules, Regulatory Guides, Standard Review Plans) of a consistent and comprehensive set of shutdown decay heat removal requirements. At the outset, particular attention will need to be focused on the further development of Regulatory Guide 1.139, "Guidance for Residual Heat Removal to Achieve and Maintain Cold Shutdown," Revision 1.

• ACRS INTERFACE INFORMATION

- Task Manager briefed the full committee on Sept. 10, 1981 on the present plans and content of the Final A-45 Task Action Plan (TAP).
- ACRS Subcommittee on Decay Heat Removal Systems (DHRS) met on Sept. 8, 1981, and Task Manager provided the subcommittee with an update of the final TAP.
- Further meetings with the full committee and subcommittee on DHRS will be held as the work on A-45 progresses and certain pre-determined milestones are completed.

• TECHNICAL ASSISTANCE CONTRACTS

- Supplemented UCLA Contract (FIN-B6835) by 10K to provide direct technical assistance to Task Manager to develop a Task Action Plan for resolution of A-45.
- Implemented a technical assistance contract with BNL (FIN-A-3381) for 50K to: 1) Obtain an assessment of the contribution to risk and/or core melt probability from decay heat removal system (DHRS) failures for six specific plants analyzed in the Reactor Safety Study (WASH-1400) and in the Reactor Safety Study Methodology Application Program (RSSMAP); 2) Make a determination whether it is feasible to separate all other currently operating U.S. commercial LWR's into groups based on whether they have similar DHRS design characteristics to the six plants covered in item 1.

– Additional technical assistance contract work will be needed for Task A-45.

– See Status Summary regarding "HOLD" on Task A-45 program.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
B6835	UCLA	$10,000	$10,000
# A3381	BNL	$50,000	$50,000

• POTENTIAL PROBLEMS

Potential problem areas identified to date are:

- Obtaining sufficient NRR manpower to work on Task A-45.
- Development of appropriate reliability / interim quantitative measure for Task A-45 and translation of probabilistic results into licensing requirements.
- Program funding must be approved and obtained. For those technical assistance tasks requiring competitive contractor bidding, that part of the program could be delayed by approximately six months to one year.
- Obtaining necessary operating experience on shutdown cooling systems including the most current DHR system design information resulting from Post-TMI changes.
- Uncertainty in the quality of information that will be available from RSS MAP-NREP, and other reliability and risk assessments, on what schedule, and whether information can be extrapolated to all operating plants.
- Close coordination and cooperation needed within NRR and between NRR and RES.
- Number of plants that need to be assessed may be significantly greater than the 6 to 10 plants that will have a PRA performed.

– Time to obtain NRR approval of funding has been greater than expected by about four months (See Status Summary).

• STATUS SUMMARY

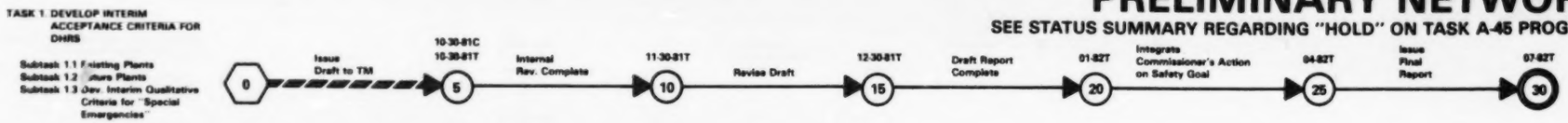
Task A-45 was approved as a USI by the NRC on 12-24-80. The task manager has been selected and a draft Task Action Plan was published on May 22, 1981.

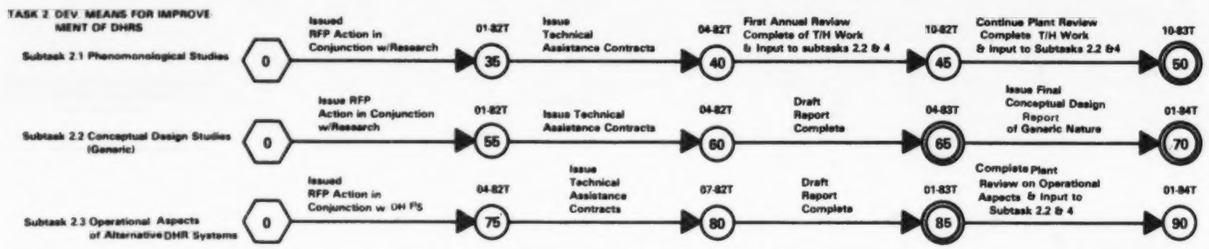
Based on internal staff and ACRS comments, a final revised TAP was approved on October 7, 1981.

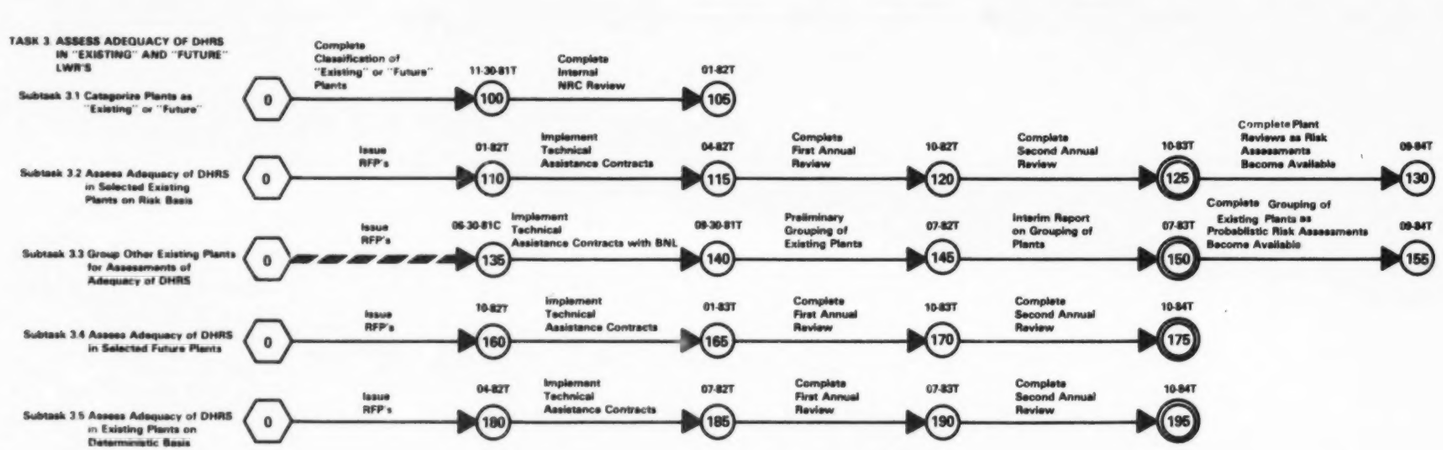
Because the Director of NRR did not approve the requested funding level for Task A-45 technical assistance work, a "HOLD" has been placed on the program. Various options have been considered for Task A-45, most of which include reduced scope, funding and completion time. A recommended option will be presented to the Director of NRR for approval. After approval has been obtained, a revised TAP will be published.

SHUTDOWN DECAY HEAT REMOVAL REQUIREMENTS (A-45)
PRELIMINARY NETWORK

SEE STATUS SUMMARY REGARDING "HOLD" ON TASK A-45 PROGRAM.

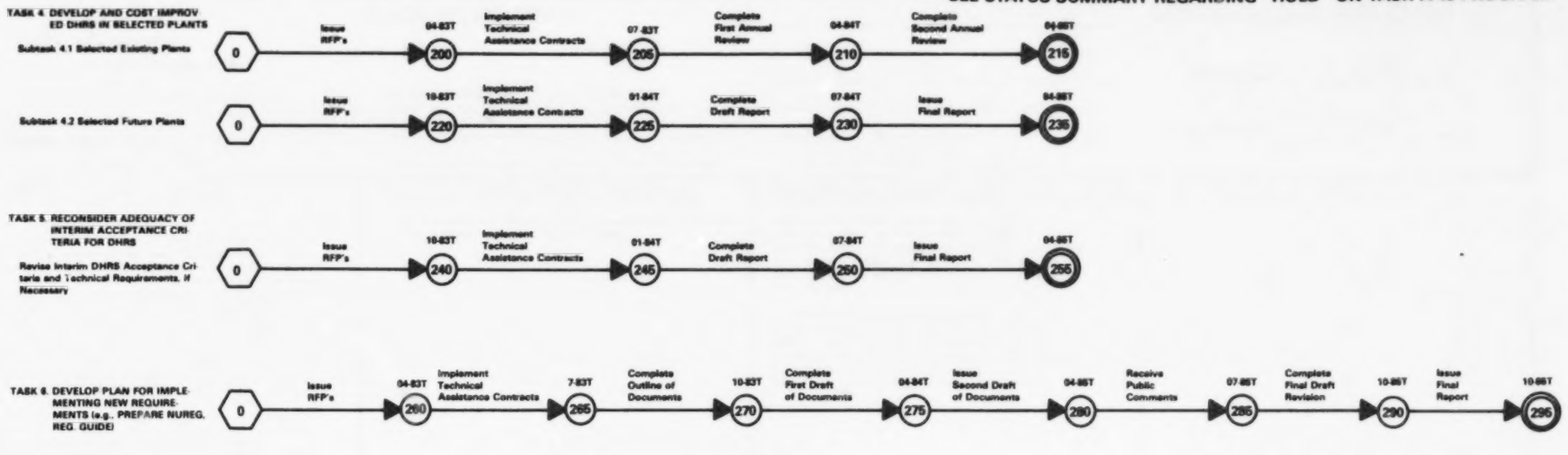






SHUTDOWN DECAY HEAT REMOVAL REQUIREMENTS (A-45) (CONTINUED)
PRELIMINARY NETWORK

SEE STATUS SUMMARY REGARDING "HOLD" ON TASK A-45 PROGRAM.



SEISMIC QUALIFICATION OF EQUIPMENT IN OPERATING PLANTS (A-46)

AS OF WEEK ENDING FEBRUARY 19, 1982

KEY PERSONNEL

TASK MANAGER
PATRICK SEARS X24216
Patrick Sears

MPA ANALYST
B. DULIN X28330

TASK REVIEWERS

NAME	BRANCH
# JERRY JACKSON	EQB/DE

SCHEDULED COMPLETION

ORIGINAL 12-15-83

CURRENT # N/S

• PROBLEM DESCRIPTION

The design criteria and methods for the seismic qualification of mechanical and electrical equipment in nuclear power plants have undergone significant change during the course of the commercial nuclear power program. Consequently, the margins of safety provided in existing equipment to resist seismically induced loads and perform the intended safety functions may vary considerably. The seismic qualification of the equipment in operating plants must, therefore, be reassessed to ensure the ability to bring the plant to a safe shutdown condition when subject to a seismic event. The objective of this Unresolved Safety Issue is to establish an explicit set of guidelines that could be used to judge the adequacy of the seismic qualification of mechanical and electrical equipment at all operating plants in lieu of attempting to backfit current design criteria for new plants. This guidance will concern equipment required to safely shutdown the plant, as well as equipment whose function is not required for safe shutdown, but whose failure could result in adverse conditions which might impair shutdown functions. Also, explicit guidelines will be established for use in requalifying equipment whose seismic qualification was found to be inadequate.

A preliminary breakdown of the tasks is as follows:

- Task 1 Develop Basis for Selecting Equipment to be Qualified.
- Task 2 Correlation of Inservice Dynamic Responses in Non-Nuclear Facilities to the Qualification of Nuclear Plant Equipment.
- Task 3 Develop Methods for Qualification of Equipment in Operating Plants Using In-Situ Test Methods.
- Task 4 Survey Existing Seismic Qualification Criteria and Procedures.
- Task 5 Develop Guidelines and Criteria for Qualification of Response Spectra.
- Task 6 Document results of Task A-46 and prepare NUREG.

• RES INTERFACE INFORMATION

RES has initiated a contract with Southwest Research Institute to survey seismic qualification methodology.

• ACRS INTERFACE INFORMATION

A preliminary Task Action Plan was presented to the ACRS. The staff plans to meet with the ACRS at appropriate times to present the status of program results.

• TECHNICAL ASSISTANCE CONTRACTS

A statement of work has been written to study appropriate methods of requalifying equipment in operating plants. Laboratory and in-situ testing are being considered. A funding authorization of $50K has been sent to Edgerton, Germeshausen & Grier (EG&G) to start work on Task 3.

A statement of work has been written to review and analyze inservice dynamic response data observed in non-nuclear power plants and industrial facilities to aid in the qualification of equipment in nuclear plants. A funding authorization of $50K has been sent to Lawrence Livermore Laboratory (LLL) to start work on Task 2.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED

Contracts are being negotiated.

• POTENTIAL PROBLEMS

• STATUS SUMMARY

This program is being redirected to ensure compatibility with Equipment Qualification Program Plan. Detailed milestone charts are under development.

# INDICATES A CHANGE IN INFORMATION

SAFETY IMPLICATIONS OF CONTROL SYSTEMS (A-47)

AS OF WEEK ENDING FEBRUARY 19, 1982

KEY PERSONNEL

TASK MANAGER
ANDREW G ZUKIEWICZ (GIB) X24713

F K Zukiewicz

MPA ANALYST
M. ANDERSON X27021

TASK REVIEWERS

NAME	BRANCH
D. BASDEKAS	RES/DFO
C. ROSSI	ICSB/DSI
J. CONRAN	RRAB/DSI
#F. ORR	RSB/DSI
M. CHIRAMAL	AEOD/PSU
J. T. BEARD	DL/ORAB

SCHEDULED COMPLETION

1978 ANNUAL REPORT	NOT DETERMINED
CURRENT	N/S

• PROBLEM DESCRIPTION

This issue concerns the potential for accidents or transients being made more severe as a result of control system failures or malfunctions. These failures or malfunctions may occur independently or as a result of the accident or transient under consideration and would be in addition to any control system failure that may have initiated the event. Although it is generally believed that control system failures are not likely to result in loss of safety functions which could lead to serious events or result in conditions that safety systems are not able to cope with, in-depth studies have not been performed to support this belief. The potential for an accident that would affect a particular control system - and the effects of the control system failures - will differ from plant to plant. Therefore, it is not likely that it will be possible to develop generic answers to these concerns, but rather plant-specific reviews will be required. The purpose of this Unresolved Safety Issue is to define generic criteria that may be used for plant-specific reviews. A specific subtask of this issue will be to study the steam generator overfill transient in PWRs and the reactor overfill transient in BWRs to determine and define the need for preventive and/or mitigating design measures to accommodate this transient.

• RES INTERFACE INFORMATION

Close coordination will be required on Task A-47 between NRR and RES. RES assistance will be required from the Division of Facilities Operations and the Division of Risk Analysis. The Division of Facilities Operations will provide input from their research programs (currently being developed in conjunction with A-47) on control system interactions. The Division of Risk Analysis will provide input on their current ongoing ORNL program ("Evaluation of Pressurized Thermal Shock"). Other research activities are being evaluated to determine their significance to A-47 and may be identified later.

• ACRS INTERFACE INFORMATION

The ACRS Subcommittee on Electrical Power Systems (Dr. Kerr) met on the issue on 1/23/81 to initiate an approximate six month study at the request of Chairman Ahearne.

Elements of conceptual Task Action Plan were discussed with ACRS on 06-07-81.

• TECHNICAL ASSISTANCE CONTRACTS

During the week of July 20, 1981, meetings were held with EG&G (Idaho Falls), Battelle Northwest and Lawrence Livermore Labs. for the purpose of discussing the A-47 objectives and to assess the type of activities that could best be conducted by these organizations. Similar meetings with ORNL and Sandia Labs. were held July 9, 1981. On November 4 and 5, 1981 meetings were held at Oak Ridge to discuss ORNL & Sandia research activities on selected Operating Reactor Design (to be selected). These activities are sponsored by NRR/RES and will be compatible with the needs of A-47. Technical Assistance Contract work will be needed for the following Tasks and will be initiated in early FY82.

1. Evaluate Control System Failures Leading to Steam Generator or Reactor Overfill Transients FIN # (not assigned).
2. Evaluate Control System Failures Leading to Reactor Overcooling Transients. FIN # (not assigned).
3. Evaluate (all other) Control Systems that may have Safety Implications FIN # (not assigned).

Additional Technical Assistance Contract work will be needed for Task A-47 and is being defined as part of the developing Task Action Plan.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
Contracts are being negotiated.			

• POTENTIAL PROBLEMS

Obtain necessary operating experience information on Non-Safety Control Systems for the Nuclear Steam System (NSS) and the Balance of Plant (BOP) systems.

Obtain necessary design information (i.e., schematics and systems descriptions) for the Non-Safety Control System designs for the NSS and the BOP designs.

Traditionally, licensees were not required to provide design or operating experience on Non-Safety Systems and therefore this information may be difficult to obtain.

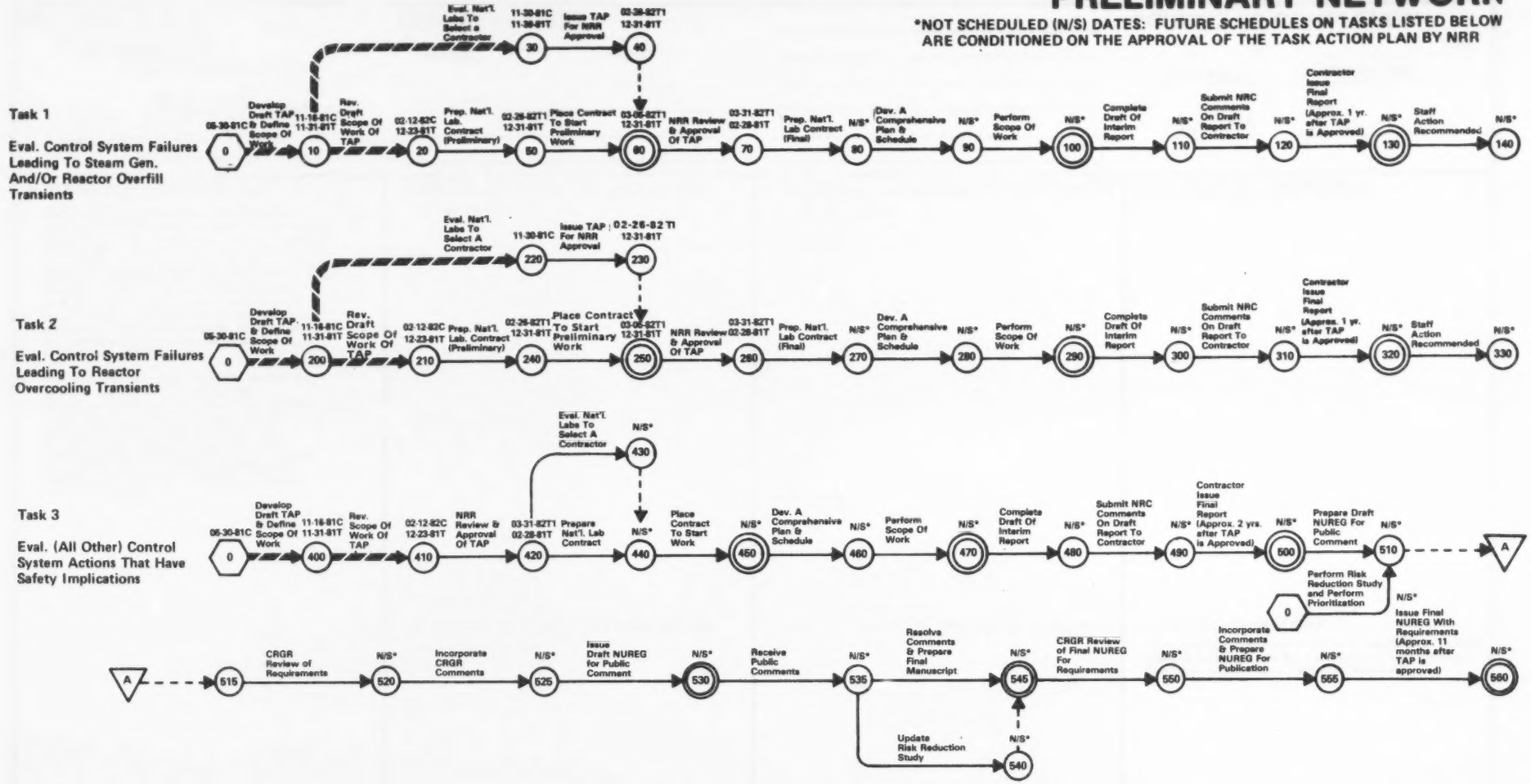
• STATUS SUMMARY

Task A-47 was approved as a USI by the NRC on 12/24/80. The task manager has been selected. A revised Draft Task Action Plan will be issued by the end of February 1982 for staff concurrence and approval. A conceptual network has been developed and the task work schedules are being developed. Management approval is expected by 04/01/82.

# INDICATES A CHANGE IN INFORMATION

SAFETY IMPLICATIONS OF CONTROL SYSTEMS (A-47)
PRELIMINARY NETWORK

*NOT SCHEDULED (N/S) DATES: FUTURE SCHEDULES ON TASKS LISTED BELOW ARE CONDITIONED ON THE APPROVAL OF THE TASK ACTION PLAN BY NRR



HYDROGEN CONTROL MEASURES AND EFFECTS OF HYDROGEN BURNS ON SAFETY EQUIPMENT (A-48)

AS OF WEEK ENDING FEBRUARY 19, 1982

KEY PERSONNEL

TASK MANAGER
CLIFFORD ANDERSON X27488

Clifford J. Anderson

MPA ANALYST
D. SCHWARZ X27106

TASK REVIEWERS

NAME	BRANCH
NOT ASSIGNED	

SCHEDULED COMPLETION

ORIGINAL: NOT DETERMINED

CURRENT: N/S

• PROBLEM DESCRIPTION

Postulated reactor accidents which result in a degraded or melted core can result in generation and release to the containment of large quantities of hydrogen. The hydrogen is formed from the reaction of the zirconium fuel cladding with steam at high temperatures and/or by radiolysis of water. Experience gained from the TMI-2 accident indicates that we may want to require more specific design provisions for handling larger hydrogen releases than currently required by the regulations particularly for smaller, low pressure containment designs.

This USI will investigate means to predict the quantity and release rate of hydrogen following degraded core accidents and various means to cope with large releases to the containment such as inerting of the containment or controlled burning. The potential effects of proposed hydrogen control measures on safety including the effects of hydrogen burns on safety related equipment will be investigated.

• RES INTERFACE INFORMATION

TO BE DEVELOPED

• ACRS INTERFACE INFORMATION

TO BE DEVELOPED

• TECHNICAL ASSISTANCE CONTRACTS

TO BE DEVELOPED

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED

• POTENTIAL PROBLEMS

• STATUS SUMMARY

Task A-48 was approved as a USI by the NRC on 12/24/80. Clifford Anderson has been assigned as task manager. The Task Action Plan is under development.

# INDICATES A CHANGE IN INFORMATION

PRESSURIZED THERMAL SHOCK (A-49)

AS OF WEEK ENDING FEBRUARY 19, 1982

KEY PERSONNEL

TASK MANAGER
ROY WOODS X24714

Roy Woods

MPA ANALYST
DIANE SCHWARZ X27106

TASK REVIEWERS

NAME	BRANCH
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A. OXFURTH	LGB/DST
L. LOIS	CPB/DSI
J. STROSNIDER	RES
M. VAGINS	RES
R. JOHNSON	GIB/DST
R. KLECKER	MTEB/DE
N. RANDALL	MTEB/DE
G. VISSING	ORB-4/DL
J. CLIFFORD	PTRB/HFS
C. JOHNSON	RES

SCHEDULED COMPLETION

ORIGINAL: Not Determined

CURRENT: N/S

• PROBLEM DESCRIPTION

Neutron irradiation of reactor pressure vessel weld and plate materials decreases the fracture toughness of the materials. The fracture toughness sensitivity to radiation induced change is increased by presence of certain materials such as copper. Decreased fracture toughness makes it more likely that, if a severe overcooling event occurs followed by or concurrent with high vessel pressure, and if a small crack is present on the vessels inner surface, that crack could grow to a size that might threaten vessel integrity.

Severe pressurized overcooling events are improbable since they require multiple failures and improper operator performance. However, certain precursor events have happened that could have potentially threatened vessel integrity if additional failures had occurred and/or if the vessel had been more highly irradiated. Therefore, the possibility of vessel failure due to a severe pressurized overcooling event cannot be ruled out.

• RES INTERFACE INFORMATION

A major portion of the work is being performed under a contract with Oak Ridge National Laboratory through the Division of Risk Analysis, RES (FIN # B0466).

Other major contributors are:

Primary System Integrity Research Program through the Division of Engineering Technology, RES and

Code Applications Program through the Division of Accident Evaluation, RES

• ACRS INTERFACE INFORMATION

Meetings have been held and will be scheduled as necessary with the Subcommittee on Metallic Components and with the full ACRS.

• TECHNICAL ASSISTANCE CONTRACTS

No definite commitments to date.

There will probably be work initiated shortly in the areas of Probability and Risk Assessment (PRA), Systems, and Human Factors.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED

• POTENTIAL PROBLEMS

PRA and Systems aspects of the effort require experienced reviewers who are in great demand and therefore difficult to obtain for this effort.

Human Factors (procedures) aspects of the effort are difficult to separate from other efforts (and such separation is not desirable in some respects). Progress is therefore slow.

• STATUS SUMMARY

Event trees/PRA/Systems Studies are in progress for the first lead plant (Oconee).

The "150 day" responses are under review.

Thermal Hydraulic Code development is underway.

# INDICATES A CHANGE IN INFORMATION

NRC FORM 335 (7 77) U.S. NUCLEAR REGULATORY COMMISSION

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16. ABSTRACT *(200 words or less)*

Provide an overview of the status of the progress and plans for resolution of the generic tasks addressing "Unresolved Safety Issues" as reported to Congress.

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UNRESOLVED SAFETY ISSUES SUMMARY

AQUA BOOK

Manuscript Completed: May 1982
Date Published: June 1982

Prepared for: OFFICE OF RESOURCE MANAGEMENT

Prepared by: OFFICE OF RESOURCE MANAGEMENT
U.S. NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

FOREWORD

THE "UNRESOLVED SAFETY ISSUES" SUMMARY IS DESIGNED TO PROVIDE THE MANAGEMENT OF THE NUCLEAR REGULATORY COMMISSION WITH A QUARTERLY OVERVIEW OF THE PROGRESS AND PLANS FOR COMPLETION OF GENERIC TASKS ADDRESSING "UNRESOLVED SAFETY ISSUES" REPORTED TO CONGRESS PURSUANT TO SECTION 210 OF THE ENERGY REORGANIZATION ACT OF 1974 AS AMENDED. THIS SUMMARY UTILIZES DATA COLLECTED FROM THE OFFICE OF NUCLEAR REACTOR REGULATION, OFFICE OF NUCLEAR REGULATORY RESEARCH, AND THE NATIONAL LABORATORIES AND IS PREPARED BY THE OFFICE OF RESOURCE MANAGEMENT.

THE DEFINITION OF WHAT CONSTITUTES COMPLETION OF AN UNRESOLVED SAFETY ISSUE (USI) HAS RECENTLY BEEN EXPANDED TO INCLUDE THE IMPLEMENTATION OF THE TECHNICAL RESOLUTION. THIS IS IN ACKNOWLEDGEMENT OF THE FACT THAT REAL SAFETY BENEFITS OCCUR ONLY AFTER THE IMPLEMENTATION HAS TAKEN PLACE. IMPORTANT ELEMENTS OF THIS IMPLEMENTATION PHASE ARE:

1. THE PROVISION OF A PUBLIC COMMENT PERIOD FOLLOWING THE ISSUANCE OF A DRAFT NUREG REPORT INCORPORATING THE STAFF'S TECHNICAL RESOLUTION FOLLOWED BY A DISCUSSION AND DISPOSITION OF THE COMMENTS RECEIVED IN A FINAL NUREG REPORT.

2. THE PROVISION FOR INCORPORATION OF THE TECHNICAL RESOLUTION INTO THE NRC'S REGULATIONS, STANDARD REVIEW PLAN, REGULATORY GUIDES, OR OTHER NRC OFFICIAL GUIDANCE OR REQUIREMENTS AS APPROPRIATE.

3. THE PROVISION FOR APPLICATION OF THE TECHNICAL RESOLUTION TO INDIVIDUAL OPERATING PLANTS IN THE FORM OF HARDWARE OR DESIGN CHANGES, TECHNICAL SPECIFICATION CHANGE, AND/OR CHANGE TO OPERATING PROCEDURES AS APPROPRIATE.

THE SCHEDULES IN THIS BOOK INCLUDE A MILESTONE AT THE END OF EACH ACTION PLAN WHICH REPRESENTS THE INITIATION OF THE IMPLEMENTATION PROCESS BOTH WITH RESPECT TO INCORPORATION OF THE TECHNICAL RESOLUTION IN THE NRC OFFICIAL GUIDANCE OR REQUIREMENTS AND ALSO THE APPLICATION OF CHANGES TO INDIVIDUAL OPERATING PLANTS. THE SCHEDULE FOR IMPLEMENTATION WILL NOT NORMALLY BE INCLUDED IN THE TASK ACTION PLAN(S) FOR THE RESOLUTION OF A USI SINCE THE NATURE AND EXTENT OF THE ACTIVITIES NECESSARY TO ACCOMPLISH THE IMPLEMENTATION CANNOT NORMALLY BE REASONABLY DETERMINED PRIOR TO THE DETERMINATION OF A TECHNICAL RESOLUTION. THE PROGRESS AND STATUS FOR IMPLEMENTATION OF UNRESOLVED SAFETY ISSUES FOR WHICH A TECHNICAL RESOLUTION HAS BEEN COMPLETED ARE REPORTED SPECIFICALLY IN A SEPARATE TABLE PROVIDED IN THIS SUMMARY.

KARL KNIEL, CHIEF OF THE GENERIC ISSUES BRANCH, DIVISION OF SAFETY TECHNOLOGY/NRR, IS RESPONSIBLE FOR MANAGING THE GENERIC TASKS INCLUDED IN THIS SUMMARY.

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ABBREVIATIONS

AAB:	Accident Analysis Branch (Former NRR Branch)
AB:	Administration Branch, Training and Administration Staff (IE)
AC:	Alternating Current
ACRS:	Advisory Committee on Reactor Safeguards
AD:	Assistant Director
ADB:	Analysis and Development Branch, Division of Reactor Safety Research (RES)
AEB:	Accident Evaluation Branch, Division of Systems Integration (NRR)
AEOD:	Office of the Analysis and Evaluation of Operational Data
AIF:	Atomic Industrial Forum
APTS:	Action Plan Tracking System
ARL:	Alden Research Laboratory
ASB:	Auxiliary Systems Branch, Division of Systems Integration (NRR)
ASME:	American Society of Mechanical Engineers
ASTM:	American Society of Testing Materials
ATWS:	Anticipated Transient Without Scram
B&W:	Babcock and Wilcox Company
BNC:	Brookhaven National Conservatory
BNL:	Brookhaven National Laboratory
BOP:	Balance of Plant
BWR:	Boiling Water Reactor
CE:	Combustion Engineering, Incorporated
CEB:	Chemical Engineering Branch, Division of Engineering (NRR)
CFR:	Code of Federal Regulations
CO:	Condensation Isolation
CP:	Construction Permit
CPB:	Core Performance Branch, Division of Systems Integration (NRR)
CR:	Contractor Report
CRGR:	Committee for Review of Generic Requirements
CSB:	Containment Systems Branch, Division of Systems Integration (NRR)
DC:	Direct Current
DE:	Division of Engineering (NRR)
DFO:	Division of Facility Operations (RES)
DHFS:	Division of Human Factors Safety (NRR)
DHRS:	Decay Heat Removal System
DL:	Division of Licensing (NRR)
DOE:	U. S. Department of Energy
DOR:	Division of Operating Reactors (Former NRR Division)
DRA:	Division of Risk Analysis (RES)
DSI:	Division of Systems Integration (NRR)
DSS:	Division of Systems Safety (Former NRR Division)
DST:	Division of Safety Technology (NRR)
E:	Engineering
EB:	Enforcement Branch, Enforcement and Investigations Staff (IE)
ECC:	Emergency Core Cooling
EEB:	Environmental Engineering Branch, Division of Engineering (NRR)
EFPY:	Effective Full-Power Years
EG&G:	Edgerton, Germeshausen & Grier
EP:	Emergency Preparedness
EPRI:	Electric Power Research Institute
EQB:	Equipment Qualification Branch, Division of Engineering (NRR)
FIN:	Financial
FIRL:	Franklin Institute Research Laboratories
FSI:	Fluid Structure Interaction
FSTF:	Full-Scale Test Facility
FW:	Feedwater
GB:	Geosciences Branch, Division of Engineering (NRR)
GE:	General Electric
GIB:	Generic Issues Branch, Division of Safety Technology (NRR)
HFEB:	Human Factors Engineering Branch, Division of Health, Siting and Waste Mgmt. (RES)
HFS:	Division of Human Factors Safety (NRR)
HSST:	Heavy Section Steel Technology
ICBR:	Instrumentation and Control Branch, Division of Facility Operations (RES)
ICSB:	Instrumentation and Control Systems Branch, Division of Systems Integration (NRR)
IE:	Office of Inspection and Enforcement
IEEE:	Institute of Electrical and Electronics Engineers
INEL:	Idaho Nuclear Engineering Laboratory
IREP:	Integrated Reliability Evaluation Program
ISI:	In-Service Inspection
LER:	Licensee Event Report
LGB:	Licensing Guidance Branch, Division of Safety Technology (NRR)
LLL:	Lawrence Livermore Laboratory
LOCA:	Loss-of-Coolant Accident
LPP:	Lead Plant Program
LTP:	Long Term Program
LWR:	Light-Water Reactor
MARK I-III:	Containment Types for Boiling Water Reactors
MEB:	Mechanical Engineering Branch, Division of Engineering (NRR)
MEBR:	Materials Engineering Branch, Division of Engineering Technology (RES)
MIT:	Massachusetts Institute of Technology
MMRB:	Metallurgy and Materials Branch, Division of Reactor Safety Research (RES)
MSLB:	Main Steam Line Break
MTEB:	Materials Engineering Branch, Division of Engineering (NRR)
NDE:	Non-Destructive Examination
NREP:	Neutron Resonance Escape Probability
NRL:	Naval Research Laboratory
NRR:	Office of Nuclear Reactor Regulation
NSS:	Nuclear Steam System
NUREG:	Nuclear Regulation

ABBREVIATIONS

OG:	Owner's Group
OL:	Operating License
ORAB:	Operating Reactors Assessment Branch, Division of Licensing (NRR)
ORB:	Operating Reactors Branch, Division of Licensing (NRR)
ORNL:	Oak Ridge National Laboratory
OSD:	Office of Standards Development (Former NRC Office)
OTSG:	Once-Through Steam Generator
PASNY:	Power Authority of the State of New York
PNL:	Pacific Northwest Laboratory (Battelle)
PS:	Plant Systems
PSB:	Power Systems Branch, Division of Systems Integration (NRR)
PSU:	Plant Systems Unit (AEOD)
PSYB:	Plant Systems Branch (Former NRR Branch)
PTRB:	Procedures and Test Review Branch, Division of Human Factors Safety (NRR)
PWR:	Pressurized Water Reactor
Q:	Quality Control or Assurances
RAB:	Radiological Assessment Branch, Division of Systems Integration (NRR)
RCIC:	Reactor Core Isolation Cooling
REF:	Reference
RES:	Office of Nuclear Regulatory Research
RFP:	Request for Proposal
RHR:	Residual Heat Removal
RPV:	Reactor Pressure Vessel
PRA:	Probability and Risk Assessment
RRAB:	Reliability and Risk Assessments Branch (NRR)
RRRC:	Regulatory Requirements Review Committee
RS:	Reactor Safety (Former NRR Branch)
RSB:	Reactor Systems Branch, Division of Systems Integration (NRR)
RSR:	Reactor Safety Research
RSSMAP:	Reactor Safety Study Methodology Application Program
SCC:	Stress-Corrosion Cracking
SEB:	Structural Engineering Branch, Division of Engineering (NRR)
SEP:	Systematic Evaluation Program
SEPB:	Systematic Evaluation Program Branch, Division of Licensing (NRR)
SER:	Safety Evaluation Report
SG:	Steam Generator
SGWH:	Steam Generator Water Hammer
SNL:	Sandia National Laboratory
SRP:	Standard Review Plan
SRV:	Safety Relief Valve
SSE:	Safe Shutdown Earthquake
SSPB:	Standards Special Projects Branch (NRR)
STP:	Short-term Program
TAP:	Task Action Plan
TH:	Hot Leg Temperature
TM:	Task Manager
TMI:	Three Mile Island
USI:	Unresolved Safety Issue
W:	Westinghouse Electric Corporation
WH:	Water Hammer
WUTA:	Washington University Technology Associates

PROGRAM OVERVIEW

PROJECTED DATES FOR ISSUING NRC STAFF REPORTS

Key:
(1) LPP - Lead Plant Program
(2) LTP - Long Term Program
N/S - Not Scheduled

USI NO.	TITLE	FEB. 19, 1982 AQUA BOOK	CURRENT	REMARKS
A-1,	Water Hammer	June 1, 1983*	August 1, 1983*	NUREG/CR-2059 re: actual and suspected water hammer events has been submitted for publication. Quadrex has submitted a report evaluating LWR systems which have experienced water hammer, analyzed causes and developed recommendations to reduce or eliminate water hammer.
A-3, A-4, A-5,	Steam Generator Tube Integrity	February 1, 1983*	March 30, 1983*	The draft NUREG report has been reviewed by the technical branches and their comments incorporated. The report is being prepared for management review and then will be issued for public comment.
A-7,	Mark I Long Term Program	Supplement- March 31, 1982	Supplement- June 1, 1982	This issue is considered complete since the final NUREG was issued July 1980 and confirmatory orders implementing the findings in NUREG 0661 were issued on January 13, 1981 for 22 operating BWRs with Mark I containments. However, BNL has reviewed three confirmatory issues for an SER supplement and issued a draft report for NRR review. The NRR supplement is scheduled to be issued June 1, 1982.
A-11,	Reactor Vessel Materials Toughness	June 22, 1982*	October 15, 1982*	NUREG-0774 Vols. I and II has been completed and was issued for public comment in September 1981. The comment period ended November 27, 1981. Many comments were received and were incorporated as appropriate. The final NUREG is being prepared for CRGR review.

*Part or all the schedule slippage for this task reflects additions to the milestone schedules to provide for review by the Committee for Review of Generic Requirements. (See individual schedules for details.)

PROGRAM OVERVIEW

PROJECTED DATES FOR ISSUING NRC STAFF REPORTS

USI NO.	TITLE	Feb. 19, 1982 AQUA BOOK	CURRENT	REMARKS
A-12,	Steam Generator & Reactor Coolant Pump Supports	July 19, 1982*	November 15, 1982*	Meetings were held with EPRI and Industry groups in in December 1980, to resolve use of fracture mechanics methods for qualification of supports. NRC reviewer's comments on the final NUREG-0577 were not received until mid-January 1982, causing a delay in issuing the final report. Comments made by EPRI and the AIF subcommittee on material requirements have been incorporated into the final NUREG.
A-17,	Systems Interactions	See Item II.C.3 in Action Plan Tracking System	See Item II.C.3 in Action Plan Tracking System	Work planned under TAP A-17 will be performed under TMI Action Plan item II.C.3, Systems Interaction. The status of this item is reported in the quarterly updates of the TMI Action Plan Tracking System (APTS).
A-39,	SRV Pool Dynamic Loads	Mark II-June 15, 1982* Mark III-June 15, 1982*	Mark II-July 15, 1982* Mark III-July 15, 1982*	NUREG-0763 on in-plant SRV tests was completed and issued in June 1981. NUREG-0783 on Pool temperature Limits was issued on October 26, 1981. NUREG-0802 on SRV Quencher loads and the NUREG on suppression pool temperature limits have been prepared and are under CRGR review.
A-40,	Seismic Design Criteria	December 31, 1982*	May 15, 1983*	A draft report presenting staff position has been prepared and is under management review.
A-43,	Containment Emergency Sump	April 1, 1983*	July 15, 1983*	The in-containment insulation survey is complete and NUREG/CR-2403 was issued in November 1981. ARL sump hydraulic tests have been completed. The final reports on plant insulation and particulate ingestion effects are expected in May 1982. A resolution NUREG is in the final stages of preparation and will undergo management concurrence in July 1982.

*Part or all the schedule slippage for this task reflects additions to the milestone schedules to provide for review by the Committee for Review of Generic Requirements. (See individual schedules for details.)

PROGRAM OVERVIEW

PROJECTED DATES FOR ISSUING NRC STAFF REPORTS

USI NO.	TITLE	Feb. 19, 1982 AQUA BOOK	CURRENT	REMARKS
A-44,	Station Blackout	June 30, 1983*	October 15, 1983*	Technical Assistance Contracts with ORNL and Sandia have been issued. A request for information on Diesel Generator reliability was sent to all licensees in July 1981; responses from licensees have been received. Draft contractor technical reports were provided in May 1982.
A-45,	Shutdown Decay Heat Removal Requirements	N/S	September 30, 1985	A Task Action Plan was issued on October 7, 1981. Funding for technical assistance on A-45 has been approved. A "HOLD" was placed on this program in Nov. 1981. A technical assistance contract package was approved on April 9, 1982 and implemented with Sandia on May 10, 1982.
A-46,	Seismic Qualification of Equipment in Operating Plants	N/S	May 1, 1985	This program was redirected to be made compatibile with the Equipment Qualification Program Plan. The final Task Action Plan has been issued. Work has been initiated on all tasks with all contractors well into their first subtasks.
A-47,	Safety Implications of Control Systems	N/S	N/S	A revised draft Task Action Plan was issued in April 1982 for staff concurrence and approval. Management approval of the Task Action Plan is expected by June 1, 1982. Technical Assistance contracts have been established with ORNL and INEL.
A-48,	Hydrogen Control Measures and Hydrogen Burns on Safety Equipment	N/S	N/S	A Task Manager has been selected. A draft Task Action Plan was issued on April 2, 1982 for NRC comments. Work has been suspended pending management decision on the scope of the TAP.
A-49,	Pressurized Thermal Shock	N/S	May 30, 1983	A-49 was designated by the Commission as an USI on December 28, 1981. The Task Action Plan was approved and issued on March 26, 1982. The draft short term position is under preparation.

*Part or all the schedule slippage for this task reflects additions to the milestone schedules to provide for review by the Committee for Review of Generic Requirements. (See individual schedules for details.)

USI'S FOR WHICH TECHNICAL RESOLUTION IS COMPLETE

USI NO.	TITLE	DATE COMPLETED	REPORTS PUBLISHED	IMPLEMENTATION STATUS	IMPLEMENTATION TASK MANAGER/ TELEPHONE
A-2,	ASYMMETRIC BLOWDOWN LOADS ON REACTOR PRIMARY COOLANT SYSTEMS	DEC. 1980	NUREG-0609	ALL PWR PLANT ASSESSMENTS FOR ASYMMETRIC LOADS HAVE BEEN RECEIVED AND ARE CURRENTLY BEING EVALUATED BY STAFF AND CONTRACTOR EG&G. AS A BASIS FOR THIS EVALUATION, STAFF IS IMPLEMENTING CRITERIA DEVELOPED AS PART OF THE TASK ACTION PLAN A-2. THESE CRITERIA ARE INCLUDED IN NUREG-0609. B&W AND CE OWNERS' GROUPS HAVE RESPONDED TO THE QUESTIONS RAISED AS A RESULT OF OUR INITIAL REVIEW OF THEIR SUBMITTALS ON THE ASSESSMENT OF THE PRIMARY SYSTEM COMPONENTS AND SUPPORTS TO ASYMMETRIC LOCA LOADS. THESE RESPONSES ARE CURRENTLY BEING REVIEWED AND THE REVIEW IS EXPECTED TO BE COMPLETED BY NOVEMBER 1983. THE WESTINGHOUSE OWNERS GROUPS EXPECT A RESOLUTION OF THIS ISSUE ON THE BASIS OF 'LEAK BEFORE BREAK' CRITERIA. AN INTERNAL REPORT, RECOMMENDING ACCEPTANCE OF THE 'LEAK BEFORE BREAK' CONCEPT ISSUED BY THE STAFF IN FEBRUARY 1981, HAS BEEN DISCUSSED WITH THE ACRS AND IS UNDER REVIEW BY NRC MANAGEMENT.	JIM SHEA 492-7231
A-6,	MARK I SHORT TERM PROGRAM	DEC. 1977	NUREG-0408	COMPLETE - ALL PLANT UNIQUE ANALYSES AND EQUIPMENT MODIFICATIONS AS REQUIRED WERE REVIEWED AND ACCEPTED AND APPROPRIATE TECHNICAL SPECIFICATION CHANGES WERE MADE.	- - -

USI'S FOR WHICH TECHNICAL RESOLUTION IS COMPLETE

USI NO.	TITLE	DATE COMPLETED	REPORTS PUBLISHED	IMPLEMENTATION STATUS	IMPLEMENTATION TASK MANAGER/ TELEPHONE
A-7,	MARK I LONG TERM PROGRAM	JULY 1980	NUREG-0661	LICENSEES ARE IN THE PROCESS OF DESIGNING AND INSTALLING MODIFICATIONS TO MEET THE COMMISSION'S ORDER DATE FOR EACH OPERATING PLANT. THE LICENSEES ARE IN THE PROCESS OF PREPARING PLANT-UNIQUE ANALYSIS FOR SUBMITTAL TO THE STAFF FOR POST IMPLEMENTATION AUDIT REVIEW FOR COMPLIANCE WITH THE ACCEPTANCE CRITERIA CONTAINED IN APPENDIX A TO NUREG-0661. THREE LICENSEES/APPLICANTS HAVE SUBMITTED THEIR PLANT UNIQUE ANALYSIS TO DATE.	BYRON SIEGEL 492-7534
A-8,	MARK II CONTAINMENT POOL DYNAMIC LOADS	AUGUST 1981	NUREG-0808	THE REQUIREMENTS RECOMMENDED IN NUREG-0808 ARE BEING IMPLEMENTED DURING THE OPERATING LICENSE REVIEW FOR EACH PLANT WITH A MARK II CONTAINMENT. THESE REQUIREMENTS WILL ALSO BE INCLUDED AS AN ADDITION TO THE APPROPRIATE SECTION OF THE STANDARD REVIEW PLAN.	
A-9,	ATWS	SEPT. 1980	NUREG 0460, VOL. 4 46FR57521	TWO VERSIONS OF THE PROPOSED ATWS RULE HAVE BEEN ISSUED FOR COMMENT; THE STAFF VERSION AND AN ALTERNATE VERSION OF THE RULE PREPARED AT THE DIRECTION OF COMMISSIONER HENDRIE. FOLLOWING THE COMMENT PERIOD AND CONSIDERATION OF PUBLIC COMMENTS, A FINAL RULE WILL BE PREPARED FOR COMMISSION CONSIDERATION.	FRANK ROWSOME, RES 443-5913

USI'S FOR WHICH TECHNICAL RESOLUTION IS COMPLETE

USI NO.	TITLE	DATE COMPLETED	REPORTS PUBLISHED	IMPLEMENTATION STATUS	IMPLEMENTATION TASK MANAGER/ TELEPHONE
A-10,	BWR FEEDWATER NOZZLE CRACKING	NOV. 1980	NUREG-0619	RESPONSES FROM LICENSEES TO AN IMPLEMENTATION LETTER HAVE BEEN RECEIVED AND RECOMMENDED TREATMENT OF THESE RESPONSES HAVE BEEN SUBMITTED TO NRC MANAGEMENT. ADDITIONAL INFORMATION HAS BEEN REQUESTED OF LICENSEES. IMPLEMENTATION ACTION IS PROCEEDING. THIRTEEN PLANTS HAVE RECEIVED LETTERS ACCEPTING THEIR PROPOSED MODIFICATION PLANS. NINE PLANTS ARE BEING REVIEWED AND NEGOTIATIONS ARE UNDER WAY TO OBTAIN LICENSEE COMMITMENTS TO IMPLEMENT ALL NUREG-0619 GUIDANCE. LACROSSE, BIG ROCK POINT, AND DRESDEN 1 DO NOT HAVE SUSCEPTIBLE PLANT SYSTEM CONFIGURATIONS AND ARE CONSIDERED COMPLETE WITH REGARD TO THIS ACTION. HUMBOLDT BAY, BY VIRTUE OF ITS STATUS (SHUTDOWN, NO FORSEEABLE RESTART), IS ALSO CONSIDERED COMPLETE.	PAUL O'CONNOR 492-7215
A-24,	QUALIFICATION OF CLASS 1E SAFETY RELATED EQUIPMENT	AUG. 1981	NUREG-0588	THE NUREG REPORT WAS SUBMITTED TO THE PUBLISHER FOR PRINTING ON AUGUST 7, 1981. RESOLUTION OF PUBLIC COMMENTS AND ISSUANCE OF THE REVISED NUREG REPORT COMPLETES THE ACTIVITIES ON A-24. EQB HAS THE LEAD IN IMPLEMENTING THE POSITIONS IDENTIFIED IN THE REPORT.	JOSE' CALVO 492-8563
A-26,	REACTOR VESSEL PRESSURE TRANSIENT PROTECTION	SEPT. 1978	NUREG-0224	ALL PLANTS WERE REQUESTED TO PROVIDE AN OVERPRESSURE PREVENTION SYSTEM THAT WOULD BE USED WHENEVER THE PLANT WAS IN A COLD SHUTDOWN CONDITION. ALL PWRS IMPLEMENTED THEIR SYSTEMS WITH PRELIMINARY APPROVAL FROM THE NRC, AND A COMPLETE REVIEW TOOK PLACE ON A POST-IMPLEMENTATION BASIS. THE REVIEW CONSISTS OF THREE PARTS: A) ELECTRICAL ASPECTS: B) REACTOR SYSTEM ASPECTS; C) TECHNICAL SPECIFICATIONS. PLANTS REMAINING TO BE EVALUATED ARE BEING REVIEWED BY CONTRACT WITH EG&G. THIS INCLUDES NINE OF THE TWELVE REMAINING PLANTS. FOR THE OTHER THREE PLANTS, THE SER'S HAVE BEEN COMPLETED AND ARE IN THE PROCESS OF BEING ISSUED.	AL DEAGAZIO 492-8945

USI'S FOR WHICH TECHNICAL RESOLUTION IS COMPLETE

USI NO.	TITLE	DATE COMPLETED	REPORTS PUBLISHED	IMPLEMENTATION STATUS	IMPLEMENTATION TASK MANAGER/ TELEPHONE
A-31,	RESIDUAL HEAT REMOVAL REQUIREMENTS	1978	NO FORMAL REPORT SRP 5.4.7 REVISION	RRRC APPROVED IMPLEMENTATION PLAN JANUARY 31, 1978. IMPLEMENTATION FOR OPERATING REACTORS NOT COMPLETE.	TOM NOVAK 492-7817
A-36,	CONTROL OF HEAVY LOADS NEAR SPENT FUEL	JULY 1980	NUREG-0612	IMPLEMENTATION REQUIREMENTS WERE ISSUED TO ALL LICENSEES BY LETTER DATED DECEMBER 22, 1980. THE LETTER REQUESTED; INTERIM ACTIONS TO BE COMPLETED IN 90 DAYS, A PHASE I ACTION (REPORT, CONFIRMATION AND JUSTIFICATION) IN SIX MONTHS AND PHASE II (SPECIFIC REQUIREMENTS) IN NINE MONTHS. MOST LICENSEES HAVE RESPONDED TO THE DECEMBER 22, 1980 GENERIC LETTER AND THEIR RESPONSES ARE BEING EVALUATED. ABOUT NINE PLANTS HAVE REQUESTED EXTENSIONS TO THE SEPTEMBER 22, 1981 RESPONSE DATE. SEVERAL PLANTS HAVE NOT YET ADEQUATELY RESPONDED TO THE EXTENT THAT AN EVALUATION MAY BE COMPLETED.	BUD REQUA 492-7877
A-42,	PIPE CRACKS IN BOILING WATER REACTORS	JULY 1980	NUREG-0313 REV. 1	IN FEBRUARY 1981, NUREG-0313, REV. 1 WAS ISSUED TO ALL HOLDERS OF BWR OPERATING LICENSES OR CONSTRUCTION PERMITS AND TO ALL APPLICANTS FOR OPERATING LICENSES. BY JULY 1, 1981, THE APPLICANTS/LICENSEES WERE TO PROVIDE THEIR PROGRAM FOR REPLACEMENT OF SERVICE SENSITITIVE LINES AND WELDS, THEIR PROGRAM FOR AUGMENTED INSERVICE INSPECTION, THEIR PROGRAM FOR IMPROVING THE WATER CHEMISTRY ENVIRONMENT AND INCOPORATION OF ADEQUATE LEAK DETECTION CAPABILITY. ALL LICENSEES BUT ONE HAVE RESPONDED AND THEIR REPLIES ARE BEING EVALUATED BY A TECHNICAL SUPPORT CONTRACTOR (INEL). INITIAL TER'S ARE SCHEDULED TO BE COMPLETED IN THE FIRST AND SECOND QUARTERS OF FY83.	DICK CLARK 492-7162

WATER HAMMER (A-1)

AS OF WEEK ENDING MAY 21, 1982

KEY PERSONNEL

TASK MANAGER
ALECK SERKIZ (DST) X24217

MPA ANALYST
B. ST. MARY X28330

TASK REVIEWERS

NAME	BRANCH
S. MCKAY	PTRB/DHFS
C. GRAVES	RSB/DSI

SCHEDULED COMPLETION

1978 ANNUAL REPORT 12-31-80

CURRENT # 08-01-83

• PROBLEM DESCRIPTION

#Since 1969 there have been over 150 incidents involving water hammers in BWRs and PWRs reported. The water hammers (or steam hammers) have involved steam generator feedrings and piping, the RHR system, ECC systems, and containment spray, service water, feedwater and steam lines. The incidents have been attributed to such causes as rapid condensation of steam pockets, steam-driven slugs of water, pump startup with partially empty lines, and rapid valve motion. Most of the damage reported has been relatively minor, involving pipe hangers and restraints; however, there have been several incidents which have resulted in piping and valve damage.

Task Action Plan, TAP A-1, subdivides resolution of this Unresolved Safety Issue (USI) into the following tasks:

Task 1: Water Hammer Summary Reports

Task 2: Resolution of Water Hammer Issue

Task 3: Water Hammer Position for Operating Reactors

Task 4: Water Hammer Safety Studies

Task 1 and 4 have been completed and the reports are on the network charts which follow. The near-term position for operating reactors is to permit continued operation since, as noted above, damage attributed to water hammer has been relatively minor. The long-term aspect of Task 3, is an integral part of the resolution of this issue.

Task 2 represents the resolution aspect of this USI and will result in the final resolution.

• RES INTERFACE INFORMATION

None.

• ACRS INTERFACE INFORMATION

A presentation was made to the ECCS Subcommittee on 12/03/81, plans for resolution of A-1 were presented.

• TECHNICAL ASSISTANCE CONTRACTS

Resolution efforts for A-1 are being conducted under FIN No. A6451 at INEL. Both EG&G-Idaho and Quadrex staff are being utilized.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
FY 81:			
A6451	EG&G-Idaho	$120,000	$ 39,355
FY 82:			
A6451	EG&G-Idaho		
	10/81	$120,000	$ 44,069
	11/81	146,000	56,285
#	12/81	305,000	67,613
#	01/82	305,000	137,194
#	02/82	305,000	176,590
#	03/82	305,000	207,411

• POTENTIAL PROBLEMS

#Risk analyses and cost benefit/impact evaluations required for CRGR review packages have become pacing items since these efforts will require subcontracted external efforts. Potential of a 1-2 month schedule slippage exists.

• STATUS SUMMARY

The concluding efforts and scheduled milestones for resolution of A-1 are shown in the attached schedule. Resolution will be based on evaluations of plant systems and operating procedures, severity of events experienced and attendant safety implications. It is postulated that guidelines and procedures will be developed which will minimize, or mitigate, water hammer events. The frequency of water hammer events reported has been low, and the incurred damage principally confined to pipe hangers and snubbers.

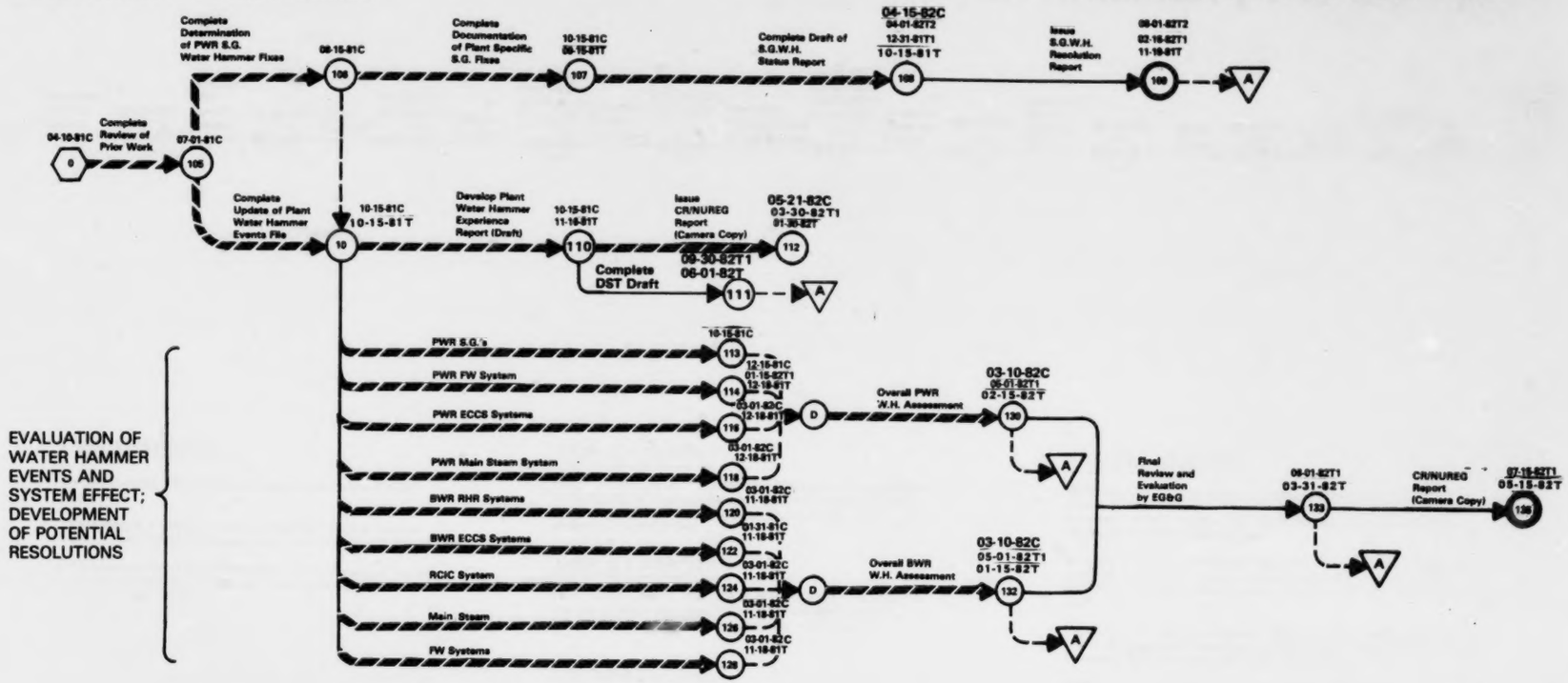
#NUREG/CR-2059, "Compilation of Actual and Suspected Water Hammer Events in Nuclear Power Plants," has been submitted to the NRC publications staff for issuance in May 1982 (see milestone #112, Task 2.0).

#Quadrex has completed an evaluation of LWR systems which have experienced water hammer, analyzed causes (i.e., faulty design vs. operational misalignments) and developed a series of recommendations to reduce or eliminate water hammer. Initial NRC staff comments have been provided to Quadrex; EG&G-Idaho staff are reviewing this report further (see milestones #130 and 132, Task 2.0).

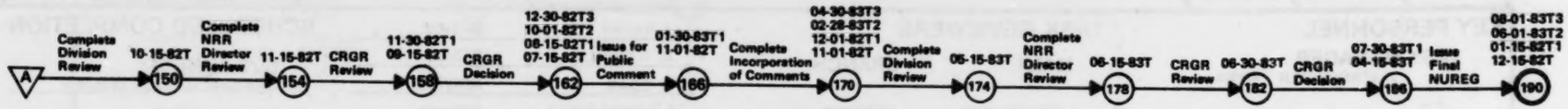
#Initial activities related to developing a technical resolution to A-1, Water Hammer are underway.

TASK 2.0
RESOLUTION OF A-1, WATER HAMMER

WATER HAMMER (A-1)



WATER HAMMER (A-1) Continued



TASK 1 REPORTS

#NUREG-0582, "Water Hammer in Nuclear Power Plants" July 1979*

Chapman, R.L., "Water Hammer Studies," EG&G Interim Rpt., CAAP-TR-053 (Rev. 1), July 1980 (Prepared for NRC Use)

#*Superceded by NUREG/CR-2059, "Compilation of Actual and Suspected Water Hammer Events in Nuclear Power Plants," May 1982.

TASK 4.1 REPORT

Chapman, R.L., et al., "Review and Evaluation of Actual and Potential Water Hammer Events in Nuclear Plants," EG&G Interim Rpt., CAAP-TR-042 (Rev. 1), September 1979 (Prepared for NRC Use)

TASK 4.2 REPORT

Watkins, J.C., Berry, R.A., "A State-of-the-Art Literature Review of Water Hammer," EG&G Interm Interim Rpt., RE-A-70-044, April 1979 (Prepared for NRC Use)

TASK 4.3 REPORT

Saha, P., et al., "An Evaluation of Condensation - Induced Water Hammer in Preheat Steam Generators, "NUREG/CR-1606, September 1980

TASK 4.4 REPORTS

Williamson, R.L., "An Analysis Tool for Predicting the Transient Hydrodynamics Resulting from the Rapid Filling of Voided Piping Systems," EG&G Interim Rpt. RE-E-79-009, February 1979 (Prepared for NRC Internal Use)

Morton, D.K., "An Analytical Procedure for Performing Structural Analyses of Nuclear Piping Systems Subjected to Fluid Transients," EG&G Interim Rpt., RE-E-79-013, February (Prepared for NRC Internal Use)

Demmie, P.N., "An Investigation of the Steam Void Collapse Water Hammer Initiating Mechanism," EG&G Interim Rpt., RE-A-78-229, February 1979 (Prepared for NRC Internal Use)

Berry, R.A., "An Analysis Tool for Predicting Transient Hydrodynamics in Nuclear Piping Systems Containing Swing Check Valves," EG&G Rpt. RE-A-78-261 (Rev. 2), September 1979 (Prepared for NRC Internal Use)

TASK 4.5 REPORT

Saffell, B.F., "Supplemental Water Hammer Analysis and System Review," EG&G preliminary Rpt. EGG-CAAP-5133, July 1980 (Prepared for NRC for Use as a Preliminary or Working Document)

WEST., CE & B&W STEAM GENERATOR TUBE INTEGRITY (A-3, A-4, A-5)

AS OF WEEK ENDING MAY 21, 1982

KEY PERSONNEL

TASK MANAGER
J. STROSNIDER X28803

P. Norian for J. Strosnider

MPA ANALYST
B. ST. MARY X28330

TASK REVIEWERS

NAME	BRANCH
J. STROSNIDER	MEBR/RES
W. PASEDAG	AEB/DSI
C. PARSZEWSKI	CEB/DE
J. RAJAN	MEB/DE
B. TUROVLIN	CEB/DE
F. ODAR	ADB/RSR
F. AKSTULEWICZ	AEB/DSI
J. ZWOLINSKI	HFEB/DHFS
L. FRANK	MEBR

SCHEDULED COMPLETION

1978 ANNUAL REPORT: Early 1980

CURRENT: # 03-30-83

• PROBLEM DESCRIPTION

Pressurized water reactor steam generator tube integrity can be degraded by corrosion induced wastage, cracking, reduction in tube diameter (denting) and vibration induced fatigue cracks. The primary concern is the capability of degraded tubes to maintain their integrity during normal operation and under accident conditions (LOCA or a main steam line break) with adequate safety margins.

Westinghouse and Combustion Engineering steam generator tubes have suffered degradation due to wastage and stress corrosion cracking. Both types of degradation have been decreased by conversion from phosphate to an all-volatile secondary water treatment. Degradation due to denting which leads to primary side stress corrosion cracking continues to be a problem.

B&W's once-through steam generators (OTSG's) were relatively free of trouble prior to the first tube leak incident at Oconee Unit 3 in July, 1976. Since then, all three Oconee units have experienced tube leak incidents. The leaks at the Oconee units are the result of cracks of unknown origins propagated in the circumferential direction by flow induced vibration and have been limited to tubes located adjacent to the open tube inspection lane.

A second form of degradation characterized as an erosion-cavitation phenomena has been observed at Oconee and other B&W units.

• RES INTERFACE INFORMATION

A. RES has funded, at the request of NRR, a major confirmatory program at PNL. The activity of this program consists of tests to verify the burst and cyclic strengths of degraded steam generator tubes and the leak rate data.
B. RES is funding a program addressing the factors which determine Inconel 600 susceptability to stress corrosion cracking in primary water. Metal condition, chemistry, temperature, stress and environment will be considered.

• ACRS INTERFACE INFORMATION

NONE

• TECHNICAL ASSISTANCE CONTRACTS

The following technical assistance contracts are generic in nature and will be applicable to the three Category "A" Technical Activities (A-3, A-4, and A-5) related to PWR steam generators.

A SANDIA – Provide statistical analysis of steam generator tube failures in operating reactors in order to establish the bases for the sampling plan for inservice inspection. Completed.
B BNL – Provide necessary computer code and perform parametric evaluation of effects of tube failures concurrent with MSLB. Completed.
C BNL – Provide technical consultation and assistance to review information in areas of water chemistory and corrosion analysis, stress and/or burst strength calculations. Completed.
D PNL – Provide cost/benefit evaluation of ISI plans. Completed.
E PNL – Evaluate environmental consequences of multiple tube failures concurrent with MSLB. Completed.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
2314	PNL	$75,000	$75,000
B2315	PNL	$95,000	$95,000

• POTENTIAL PROBLEMS

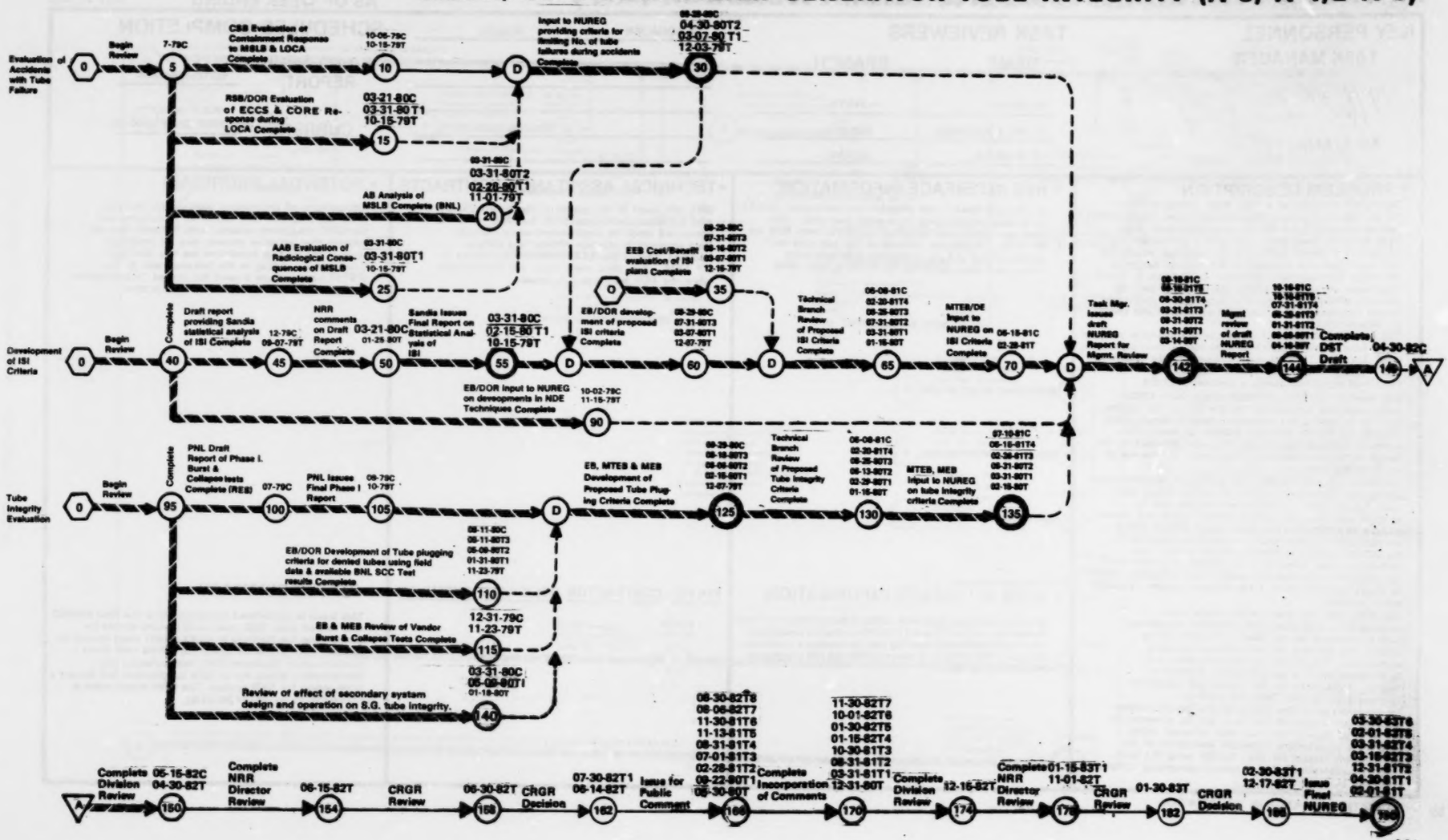
Management review and concurrence and CRGR approval are the only remaining actions necessary for issuance of the "For Comment" NUREG report.

• STATUS SUMMARY

The "For Comment" NUREG report is being prepared for submittal to CRGR.

# INDICATES A CHANGE IN INFORMATION

WEST., CE & B&W STEAM GENERATOR TUBE INTEGRITY (A-3, A-4,& A-5)



MARK I CONTAINMENT LONG-TERM PROGRAM (A-7)

AS OF WEEK ENDING MAY 21, 1982

KEY PERSONNEL

TASK MANAGER
BYRON SIEGEL X27534

Clifford Custer (acting)

MPA ANALYST
B. ST. MARY X28330

TASK REVIEWERS

NAME	BRANCH
B. SIEGEL	ORB/DL
T. SU, J. KUDRICK	CSB/DSI
R. BOSNAK	MEB/DE
F. SCHAUER	SEB/DE

SCHEDULED COMPLETION

1978 ANNUAL REPORT OCTOBER 1979

CURRENT JULY 1980C
#SUPPLEMENT 06-01-82

• PROBLEM DESCRIPTION

During the conduct of a large scale testing program for an advanced design pressure-suppression containment system (Mark III) for BWRs, new suppression pool hydrodynamic loads associated with a postulated loss of coolant accident (LOCA) were identified which had not been explicitly included in the original design of the Mark I containment systems. In addition, experience at operating plants has indicated that the dynamic effects of safety-relief valve (SRV) discharges to the suppression pool could be substantial and should be reconsidered.

The results of the Mark I containment short-term program (STP) have provided assurance that for a "most probable load" the Mark I containment system of each operating BWR facility would maintain its integrity and functional capability during a postulated LOCA.

The need exists (1) to establish design basis LOCA loads which are appropriate for the life of the facility, and (2) to restore the originally-intended design safety margins for the containment systems or ensure that adequate design safety margin has been provided in the design of the containment system prior to issuance of an operating license.

The Mark I Owner's Group has initiated a comprehensive testing and evaluation program to define design basis loads for the Mark I containment system and to establish structural acceptance criteria which will assure margins of safety for the containment system which are equivalent to that which is currently specified in the ASME Boiler and Pressure Vessel Code. Also included in their program is an evaluation of the need for structural modifications and/or load mitigation devices to assure adequate Mark I containment system structural safety margins.

The NRC staff will evaluate the loads, load combinations, and associated structural acceptance criteria proposed by the Mark I Owner's Group prior to the conduct of plant-unique structural evaluations. The results of this evaluation will be documented in a generic Safety Evaluation Report. Publication of this report will constitute the resolution of this Technical Activity.

Implementation of the results of this generic review, although not a part of this task, will be accomplished by an NRC requirement that each affected utility perform a plant-unique structural evaluation of the containment system for their facility using the loads, loading combinations, and structural acceptance criteria approved by the NRC staff.

• RES INTERFACE INFORMATION

A 1/5 scale Mark I test program was conducted by LLL to provide confirmatory pool swell load data. The final test report was submitted to PSYB from RES on 03-10-78. Additional analyses of the test data were conducted by LLL to investigate 3-D pool swell effects and were submitted to PSYB in May 1979.

• ACRS INTERFACE INFORMATION

Meetings have been held with ACRS Subcommittee on Fluid Dynamics on 9/14/79 and 11/16/79 to discuss acceptance criteria. An additional meeting may be necessary to discuss changes to the criteria. A meeting with the full committee was held on 02-06-80.

• TECHNICAL ASSISTANCE CONTRACTS

BNL will assist in the review of the hydrodynamic load definition techniques (FIN A-3108). LLL has performed work on containment safety margins, fluid structure interaction effects, and column buckling. All of the LLL work was completed in FY 1979. The BNL work scope will be complete when the SER supplement is issued.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED

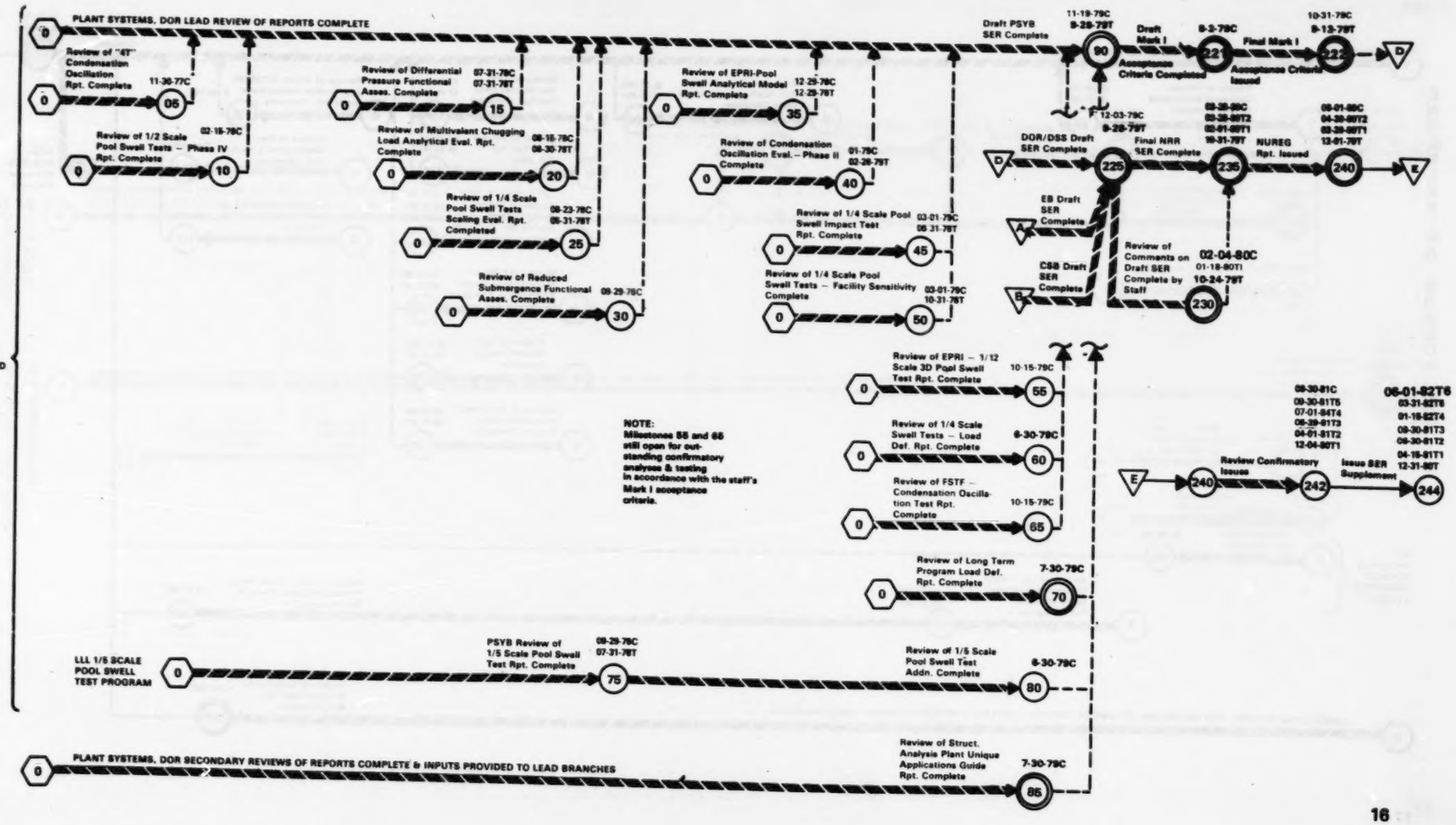
• POTENTIAL PROBLEMS

Comparisons of preliminary plant-unique analyses, using the proposed load definition techniques and the NRC acceptance criteria, with full-scale test data indicate that excessive stresses may be predicted because the loading conditions are too idealized. Confirmatory testing in the full-scale test facility (FSTF) may indicate a need for revised "condensation oscillation" loads, although none is expected.

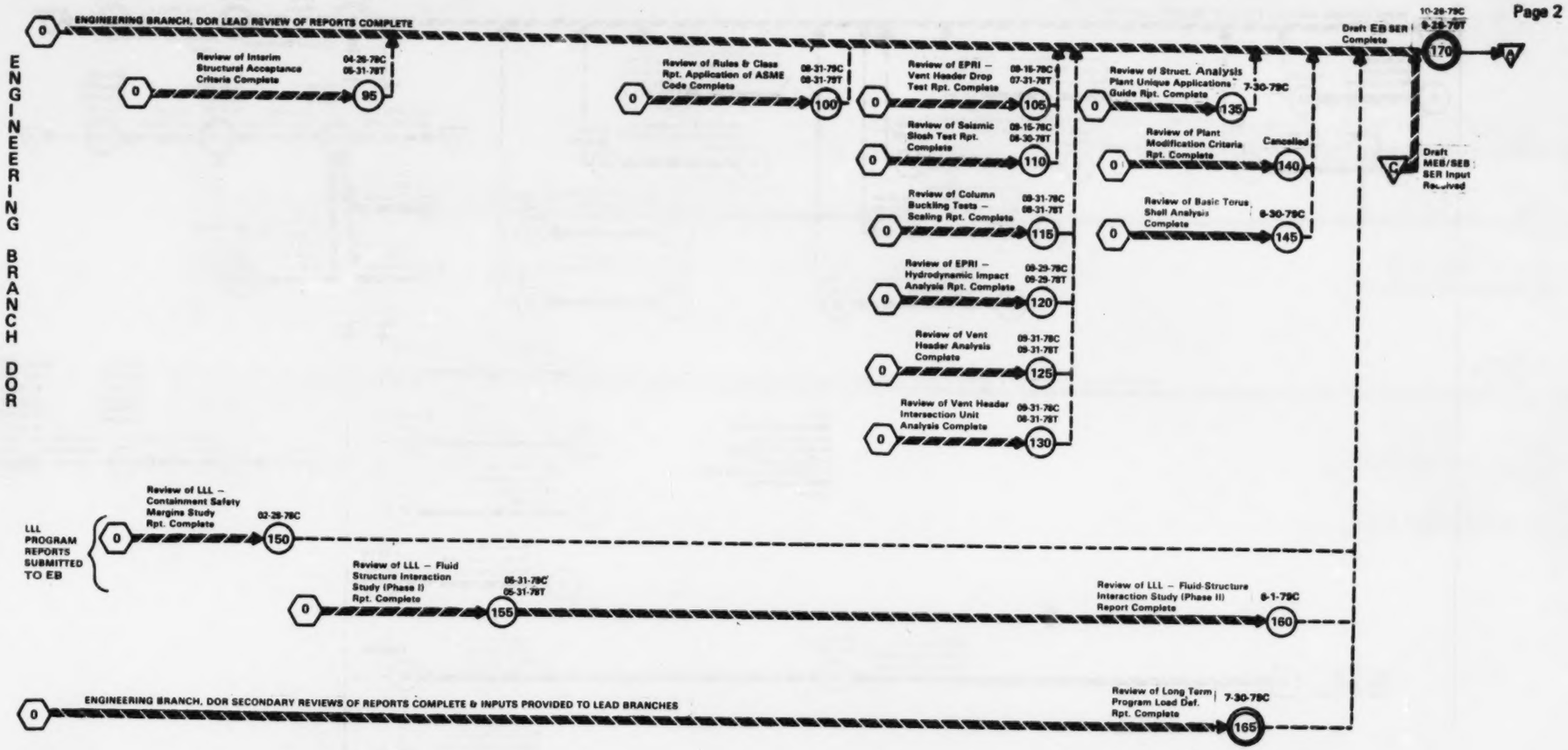
• STATUS SUMMARY

This issue is considered complete since the final NUREG was issued July, 1980, and confirmatory orders implementing the findings in NUREG-0661 were issued on January 13, 1981 for 22 operating BWRs with Mark I containments. However, BNL has reviewed three confirmatory issues for an SER supplement and issued a draft report for NRR review. The NRR supplement is # scheduled to be issued 06-01-82.

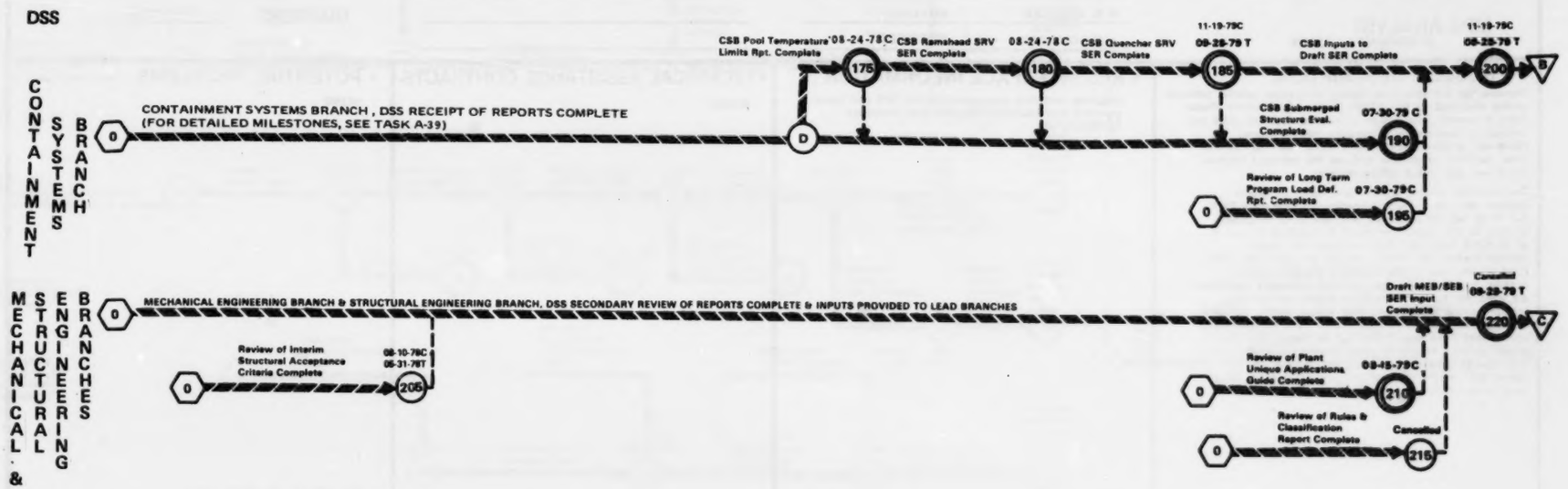
MARK I CONTAINMENT LONG-TERM PROGRAM (A-7)



MARK I CONTAINMENT LONG-TERM PROGRAM (A-7)



MARK I CONTAINMENT LONG-TERM PROGRAM (A-7)



REACTOR VESSEL MATERIALS TOUGHNESS (A-11)

AS OF WEEK ENDING May 21, 1982

KEY PERSONNEL

TASK MANAGER
RICHARD E. JOHNSON X24715

Richard E. Johnson

MPA ANALYST
D. SCHWARZ X27108

TASK REVIEWERS

NAME	BRANCH
W.S. HAZELTON	MTEB/DE
R.W. KLECKER	MTEB/DE
J. STROSNIDER	MEBR/RES
P.N. RANDALL	MEBR/RES
C.S. HINSON	RAB/DSI
C. SERPAN	MEBR/RES

SCHEDULED COMPLETION

1978 ANNUAL REPORT — July 1979

CURRENT — #10-15-82

• PROBLEM DESCRIPTION

Because the possibility of failure of nuclear reactor pressure vessels designed to the ASME Boiler and Pressure Vessel Code is remote, the design of nuclear facilities does not provide protection against reactor vessel failure. However, as plants accumulate more and more service time, neutron irradiation reduces the material fracture toughness and initial safety margins.

Results from reactor vessel surveillance programs indicate that up to approximately 20 operating PWRs will have beltline materials with marginal toughness, relative to the requirements of Appendices G and H of 10 CFR Part 20, after comparatively short (approximately 10 EFPY) periods of operation. For most plants now in the licensing process, current criteria, together with the materials currently employed, are adequate to ensure suitable safety margins for reactor vessels throughout their design lives. However, a few plants under licensing review have reactor vessels that have been identified as having the potential for marginal fracture toughness within their design lives; these vessels will have to be reevaluated in the light of the new criteria for long term acceptability.

• RES INTERFACE INFORMATION

Fracture test data was provided by RES and incorporated into NUREG-0744; RES has reviewed NUREG-0744.

• ACRS INTERFACE INFORMATION

Copies of NUREG-0744 were forwarded to the ACRS when issued for public comment; no other action planned.

• TECHNICAL ASSISTANCE CONTRACTS

NONE

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
B8486	W.U.T.A.	$103,000	$ 83,088
B0737	ORNL	$415,000	$331,375
A8419	INEL	$100,000	$ 88,740
B8809	NRL	$ 95,650	$ 95,650

• POTENTIAL PROBLEMS

NONE

• STATUS SUMMARY

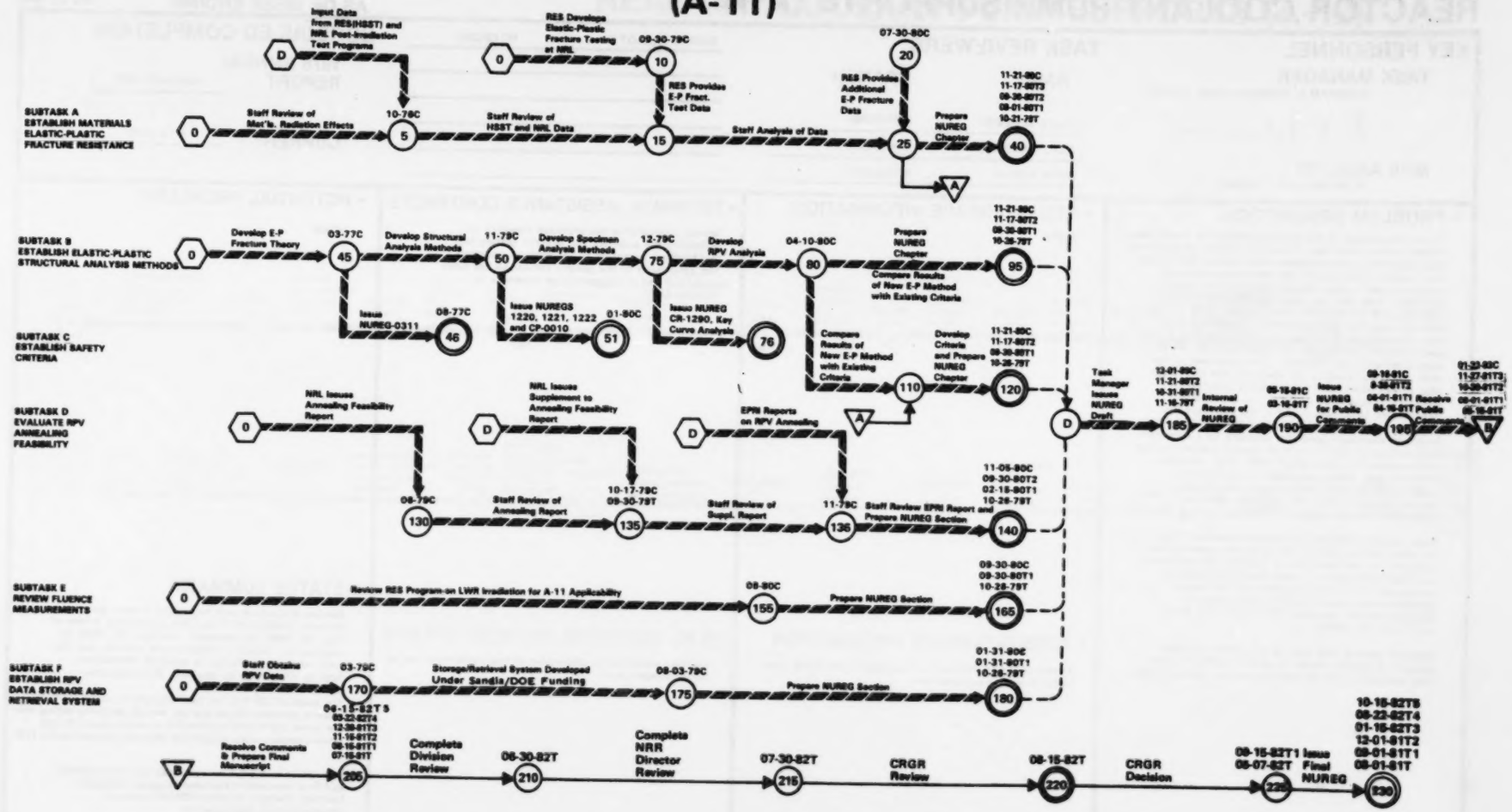
NUREG-0744 has been completed and was issued for public comment in September 1981.

Volume 1 is a summary description of the resolution and results. Volume 2 presents a detailed description of the technical work accomplished.

#Comments on NUREG-0744 have been received and incorporated as appropriate and the final NUREG is being prepared for CRGR review. The NRC has suggested to the ASME Section XI Working Group that they consider establishing safety margins to be used for calculating fracture toughness. The issue of safety margins will not be addressed in the A-11 final NUREG.

#The CRGR review schedule has been revised which results in additional schedule slippage.

REACTOR VESSEL MATERIALS TOUGHNESS (A-11)



FRACTURE TOUGHNESS OF STEAM GENERATOR AND REACTOR COOLANT PUMP SUPPORTS (A-12)

AS OF WEEK ENDING May 21, 1982

KEY PERSONNEL

TASK MANAGER
RICHARD E. JOHNSON (GIB) X24715

R. E. Johnson

MPA ANALYST
M. ANDERSON X27021

TASK REVIEWERS

NAME	BRANCH
DAVE SELLERS	MTEB(DE)
NEIL RANDALL	MTEB(DE)
DAVE SMITH	MTEB(DE)
BARRY ELLIOT	MTEB(DE)

SCHEDULED COMPLETION

1978 ANNUAL REPORT — AUGUST 1979

CURRENT — #11-15-82

• PROBLEM DESCRIPTION

During the course of the licensing action for North Anna Power Station Unit Nos. 1 and 2 a number of questions were raised as to the potential for lamellar tearing and low fracture toughness of the steam generator and reactor coolant pump support materials for those facilities. Two different steel specifications (ASTM A36-70a and ASTM A572-70a) covered most of the material used for these supports. Toughness tests, not originally specified and not in the relevant ASTM specifications, were made on those heats for which excess material was available. The toughness of the A36 steel was found to be adequate, but the toughness of the A572 steel was relatively poor at an operating temperature of 80°F. In the case of North Anna Unit Nos. 1 and 2, the applicant has agreed to raise the temperature of the ASTM A572 beams in the steam generator supports to a minimum temperature of 225°F prior to reactor coolant system pressurization to levels above 1000 psig. Auxiliary electrical heat will be supplied as necessary to supplement the heat derived from the reactor coolant loop to obtain the required operating temperature of the support materials.

Since similar materials and designs have been used on other nuclear plants, the concerns regarding the supports for the North Anna facilities are applicable to other PWR plants. It was therefore necessary to reassess the fracture toughness of the steam generator and reactor coolant pump support materials for all operating PWR plants and those in CP and OL review.

Lamellar tearing may also be a problem in those support structures similar in design to North Anna. This possibility will be investigated on a generic basis.

• RES INTERFACE INFORMATION

NONE

• ACRS INTERFACE INFORMATION

Copies of NUREG-0577 will be forwarded to the ACRS when issued in revised and final form.

• TECHNICAL ASSISTANCE CONTRACTS

Sandia Laboratories has provided expertise in evaluating the potential for lamellar tearing and low fracture toughness of the support materials. The final input to the generic resolution has been submitted and is included as an appendix to NUREG-0577.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED

Contract assistance for preparation of final report from FIRL is being managed by DL.

• POTENTIAL PROBLEMS

NONE

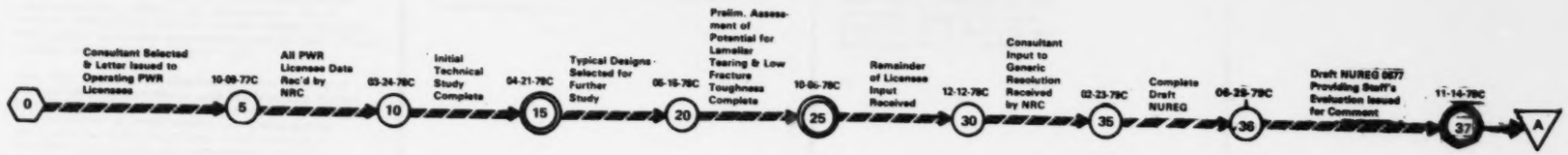
• STATUS SUMMARY

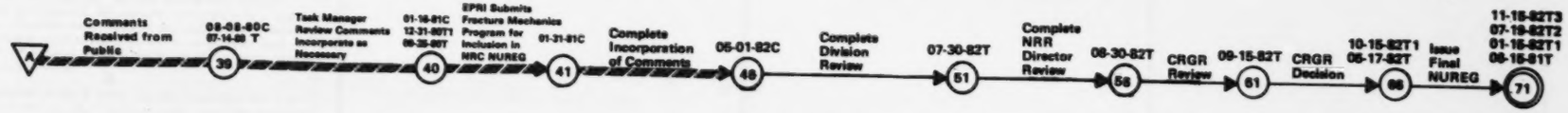
Review procedures and evaluation criteria were set forth in NUREG-0577 and the augmenting letters of May 19, 1980 (to licensees), and May 20, 1980 (to applicants). Industry response was adverse, and the staff has been requested to consider alternatives, first by EPRI and by the AIF Subcommittee on Material Requirements.

The scope of USI A-12 has been changed to reflect staff and industry concerns. Major changes include removing bolting from the scope and limiting applicability to steam generator and primary coolant pump supports as originally specified in the TAP.

#Comments have been incorporated into final NUREG and final manuscript has been prepared. Schedule has been changed to reflect CRGR review of implementation requirements.

FRACTURE TOUGHNESS OF STEAM GENERATOR AND REACTOR COOLANT PUMP SUPPORTS (A-12)





SYSTEMS INTERACTIONS IN NUCLEAR POWER PLANTS (A-17)

AS OF WEEK ENDING MAY 21, 1982

KEY PERSONNEL

TASK MANAGER
PAUL NORIAN (Acting) XZ7487

Paul Norian

MPA ANALYST
M. ANDERSON XZ7021

TASK REVIEWERS

NAME	BRANCH
NO ASSIGNED REVIEWERS	

SCHEDULED COMPLETION

1978 ANNUAL REPORT — Phase I - 09-79

CURRENT — N/A

• PROBLEM DESCRIPTION

The design of a nuclear power plant is accomplished by groups of engineers and scientists organized into engineering disciplines and into scientific disciplines. The reviews performed by the designers include interdisciplinary reviews to assure the functional compatibility of the plant structures, systems, and components. Safety reviews and accident analyses provide further assurance that system functional requirements will be met. These reviews include failure mode analyses.

The NRC review and evaluation of safety systems is accomplished in accordance with the Standard Review Plan (SRP) which assigns primary and secondary review responsibilities to organizational units arranged by plant systems or by disciplines. Each element of the SRP is assigned to an organizational unit for primary responsibility and, where appropriate, to other units for secondary responsibilities.

Thus, the design and analyses by the plant designers, and the subsequent review and evaluation by the NRC staff take into consideration the interdisciplinary areas of concern and account for systems interaction to a large extent. Furthermore, many of our regulatory criteria are aimed at controlling the risks from systems interactions. Examples include the single failure criterion and separation criteria.

Nevertheless, there is some question regarding the interaction of various plant systems, both as to the supporting roles such systems play and as to the effect one system can have on other systems, particularly with regard to whether actions or consequences could adversely affect the presumed redundancy and independence of safety systems.

The problem to be resolved by this task is to identify where the present design, analysis, and review procedures may not acceptably account for potentially adverse systems interaction and to recommend the regulatory action that should be taken to rectify deficiencies in the procedures.

• RES INTERFACE INFORMATION

The Division of Risk Analysis has been consulted during the development and execution of this plan. The Division of Engineering Technology will participate in the development of a Regulatory Guide for Systems Interaction Analyses.

• ACRS INTERFACE INFORMATION

The review is under the cognizance of the ACRS Subcommittee on Plant Arrangements. A meeting was held 02-20-80 in Washington, D.C. with the Subcommittee which recommended a demonstration of Phase I methodology before proceeding much further.

#A meeting was held 02-26-82 with the ACRS Subcommittee on Indian Point 3 to describe the PASNY systems interaction program on Indian Point 3. On 03-09-82, the ACRS documented their judgement that the PASNY plan was responsive.

• TECHNICAL ASSISTANCE CONTRACTS

Phase I work was performed by Sandia National Laboratories under FIN # A-1113(9) by NRR for $219K and under FIN # A-1084(8) by OSD for $320K. Expenditures were $197K for FIN # A-1113(9) and $320K for FIN # A-1084(8).

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
# A0403	LLNL	$150,000	$46,000
#A3355	BNL	$ 80,000	$15,000
# B2329	PNL	$150,000	$10,000
#A1296	SANDIA	$ 27,000	$16,000
A-405	LLNL	$183,000	$129,000
A-3358	BNC	$ 80,000	$ 80,000
B-2339	PNL	$ 80,000	$ 80,000

• POTENTIAL PROBLEMS

LWRs for implementation of interim guidance have not yet been selected.

• STATUS SUMMARY

The anticipated Phase II program will not be pursued as an USI. Work originally planned under TAP A-17 will now be performed under TMI Action Plan Item II-C-3, Systems Interaction. This activity will now be managed by the Systems Interaction Section, Reliability and Risk Assessment Branch, DST. The Section Leader is F. Coffman, XZ7497 The status of this item is reported in the quarterly updates of the TMI Action Plan Tracking System (APTS).

#Commission briefing scheduled for 06/82. Interim guidance for implementation on selected plants is scheduled for completion 06/82.

# INDICATES A CHANGE IN INFORMATION

DETERMINATION OF SAFETY RELIEF VALVE (SRV) POOL DYNAMIC LOADS AND TEMPERATURE LIMITS FOR BWR CONTAINMENT (A-39)

AS OF WEEK ENDING MAY 21, 1982

KEY PERSONNEL

TASK MANAGER
T.M. SU (GIB) X27477

MPA ANALYST
M. ANDERSON X27021

TASK REVIEWERS

NAME	BRANCH

SCHEDULED COMPLETION

1978 ANNUAL REPORT October 1979

CURRENT
MARK I - 02-29-80C
MARK II - 07-15-82
MARK III - 07-15-82

• PROBLEM DESCRIPTION

Operation of BWR primary system pressure relief valves can result in hydrodynamic loads on the suppression pool retaining structures or structures located within the pool. These loads result from:

1. Initial vent clearing of a relief valve pipe; and
2. Steam quenching due to locally high pool temperature.

• RES INTERFACE INFORMATION

NONE

• ACRS INTERFACE INFORMATION

Discussions with the ACRS Subcommittee on Mark III SRV loads and Kuosheng inplant test results was presented on September 25, 1981.

• TECHNICAL ASSISTANCE CONTRACTS

BNL has completed Mark II T-Quencher and Mark III X-Quencher review and is preparing input to the NUREG report.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
A3098	BNL	FY 79 $65,000 FY 80 $50,000	
A3321	BNL	FY 80 $35,000 FY 81 $70,000	

• POTENTIAL PROBLEMS

NONE

• STATUS SUMMARY

Task A-39 Mark I effort completed in February 1980.

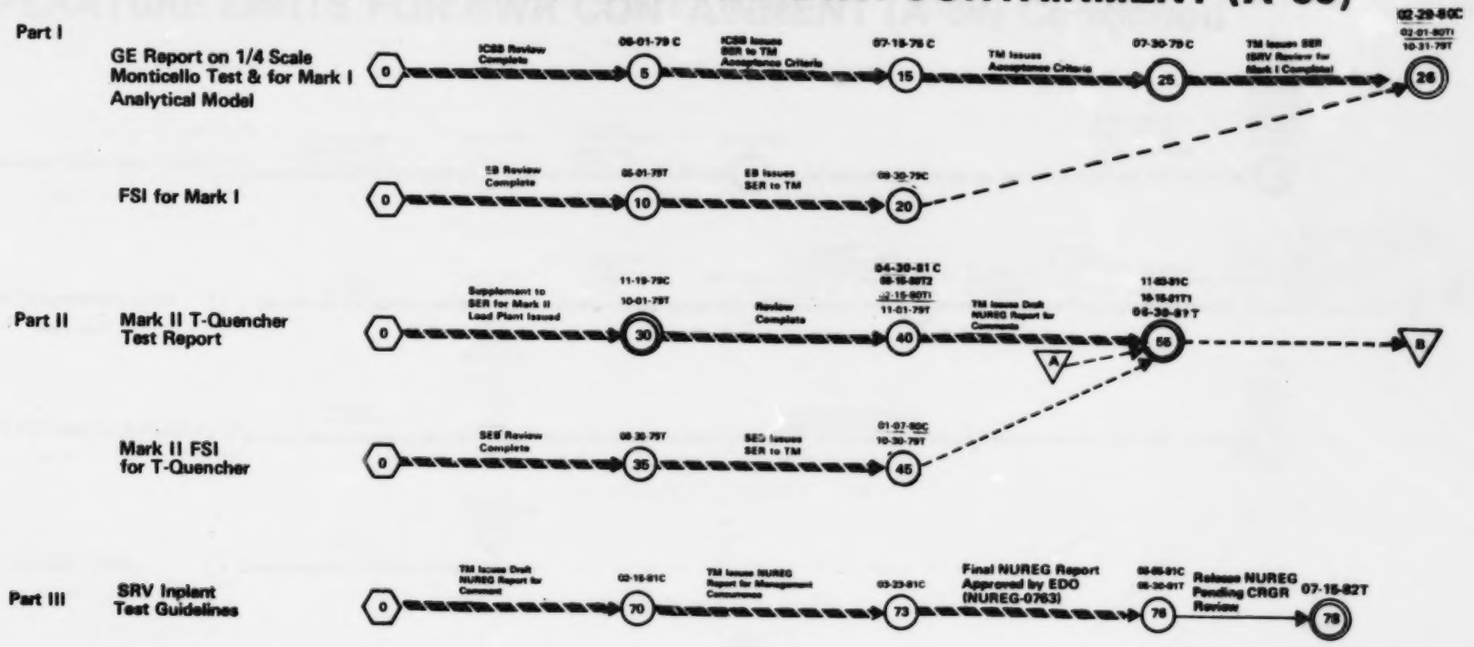
#NUREG report for inplant SRV test was approved by EDO in June 1981. Issuance of the report, however, is pending CRGR approval.

NUREG report for suppression pool temperature limit was approved by EDO in 10/81. Issuance of the #report is pending CRGR approval.

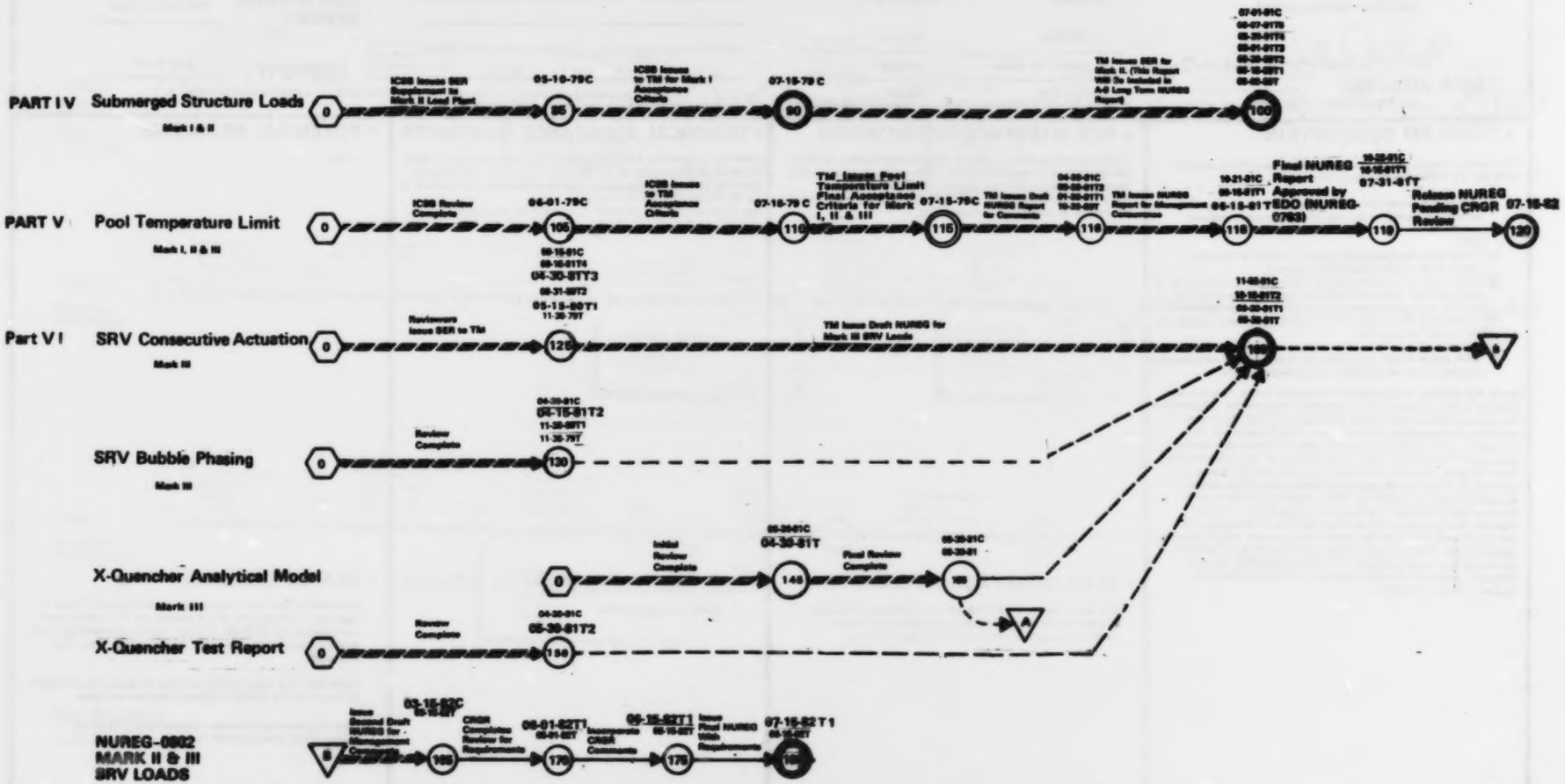
Draft NUREG report for Mark II and III SRV loads #is complete and NRR management has concurred. Submittal for CRGR review is being prepared.

# INDICATES A CHANGE IN INFORMATION

DETERMINATION OF SAFETY RELIEF VALVE (SRV) POOL DYNAMIC LOADS AND TEMPERATURE LIMITS FOR BWR CONTAINMENT (A-39)



DETERMINATION OF SAFETY RELIEF VALVE (SRV) POOL DYNAMICS LOADS AND TEMPERATURE LIMITS FOR BWR CONTAINMENT (A-39) Continued



SEISMIC DESIGN CRITERIA - SHORT TERM PROGRAM (A-40)

AS OF WEEK ENDING MAY 21, 1982

KEY PERSONNEL

TASK MANAGER
GOUTAM BAGCHI X28251

MPA ANALYST
M. ANDERSON X27021

TASK REVIEWERS

NAME	BRANCH
T. CHENG	SEB/DL
S. CHAN, H. POLK	SEB/DE
L. REITER	GB/DE

SCHEDULED COMPLETION

1978 ANNUAL REPORT — PHASE I – 1979, PHASE II – 1981

CURRENT — #05-15-83

• PROBLEM DESCRIPTION

The seismic design process required by current NRC criteria includes the following sequence of events:

(a) Define the magnitude or intensity of the earthquake which will produce the maximum vibratory ground motion at the site (the safe shutdown earthquake or SSE).
(b) Determine the free-field ground motion at the site that would result if the SSE occurred.
(c) Determine the motion of site structures by modifying the free-field motion to account for the interaction of the site structures with the underlying foundation soil.
(d) Determine the motion of the plant equipment supported by the site structures.
(e) Compare the seismic loads, in appropriate combination with other loads, on structures, systems, and components important to safety, with the allowable loads.

While this seismic design sequence includes many conservative factors, certain aspects of the sequence may not be conservative for all plant sites. At present, it is believed that the overall sequence is adequately conservative. The objective of this program is to investigate selected areas of the seismic design sequence to determine their conservatism for all types of sites, to investigate alternate approaches to parts of the design sequence, to quantify the overall conservatism of the design sequence, and to modify the NRC criteria in the Standard Review Plan if changes are found to be justified. In this manner, this program will provide additional assurance that the health and safety of the public is protected, and if possible, reduce costly design conservatisms by improving (1) current seismic design requirements, (2) NRR's capability to quantitatively assess the overall adequacy of seismic design for nuclear plants in general.

• RES INTERFACE INFORMATION

Task 1.0 will provide input to the Quantification of Conservatisms in Present Seismic Methodology, one of the three objectives of the Seismic Research Program. Close coordination with RES will be maintained through meetings to exchange information.

• ACRS INTERFACE INFORMATION

The Extreme External Phenomena Subcommittee was briefed on the Phase I reports including the recommended changes to the Seismic Design Criteria.

• TECHNICAL ASSISTANCE CONTRACTS

Lawrence Livermore Laboratory (LLL), under contract to RES, reviewed all reports by 04-30-79. LLL report on recommendations for changes to the seismic design criteria was completed on 12-28-79. (NUREG/CR-1161).
The contractor reports on tasks 7, 8, and 9 are complete.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED

Contracts complete.

• POTENTIAL PROBLEMS

None.

• STATUS SUMMARY

All contractor work necessary to complete Phase I has been completed. Staff review of LLL report was completed on 10-30-80. A draft report presenting staff position has been prepared and is under review by the staff.

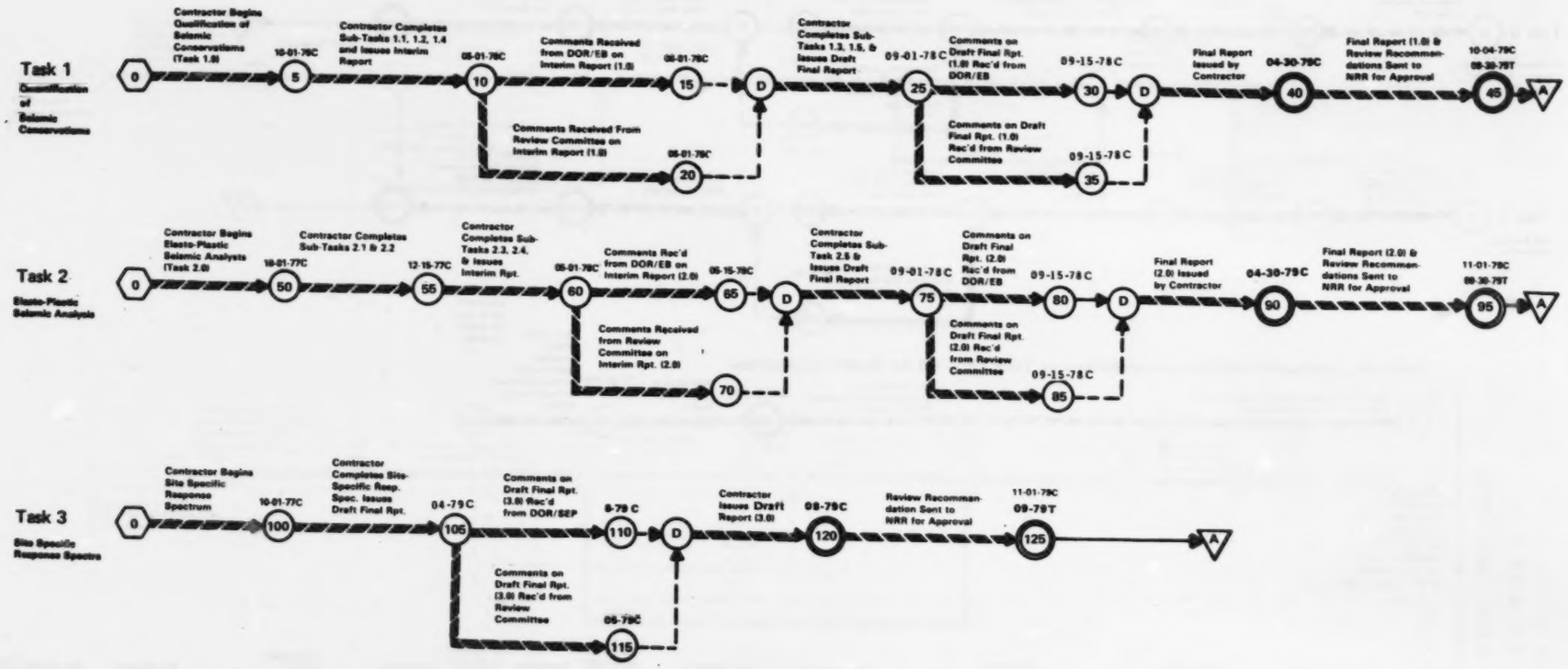
Schedule has been extended due to new requirement to incorporate CRGR implementation reviews.

Draft manuscript has been completed and is being prepared for management review.

# INDICATES A CHANGE IN INFORMATION

SEISMIC DESIGN CRITERIA - SHORT TERM PROGRAM (A-40)

Phase 1

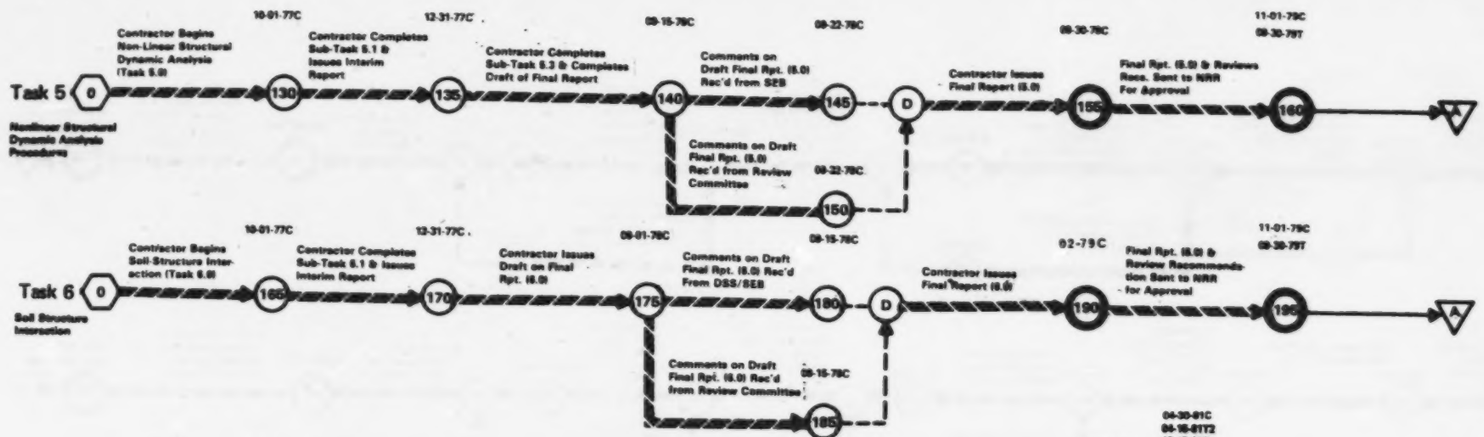

Task 1
Classification of Seismic Conservatisms
Contractor Begins Qualification of Seismic Conservatisms (Task 1.0)
10-01-79C
Contractor Completes Sub-Tasks 1.1, 1.2, 1.4 and Issues Interim Report
05-01-78C
Comments Received from DOR/EB on Interim Report (1.0)
08-01-78C
Comments Received From Review Committee on Interim Report (1.0)
08-01-78C
Contractor Completes Sub-Tasks 1.3, 1.5, & Issues Draft Final Report
09-01-78C
Comments on Draft Final Rpt. (1.0) Rec'd from DOR/EB
09-15-78C
Comments on Draft Final Rpt. (1.0) Rec'd from Review Committee
09-15-78C
Final Report Issued by Contractor
04-30-79C
Final Report (1.0) & Review Recommendations Sent to NRR for Approval
10-04-79C
09-30-79T
Task 2
Elasto-Plastic Seismic Analysis
Contractor Begins Elasto-Plastic Seismic Analysis (Task 2.0)
10-01-77C
Contractor Completes Sub-Tasks 2.1 & 2.2
12-15-77C
Contractor Completes Sub-Tasks 2.3, 2.4, & Issues Interim Rpt.
05-01-78C
Comments Rec'd from DOR/EB on Interim Report (2.0)
05-15-78C
Comments Received from Review Committee on Interim Rpt. (2.0)
Contractor Completes Sub-Task 2.5 & Issues Draft Final Report
09-01-78C
Comments on Draft Final Rpt. (2.0) Rec'd from DOR/EB
09-15-78C
Comments on Draft Final Rpt. (2.0) Rec'd from Review Committee
09-15-78C
Final Report (2.0) Issued by Contractor
04-30-79C
Final Report (2.0) & Review Recommendations Sent to NRR for Approval
11-01-78C
09-30-79T
Task 3
Site Specific Response Spectra
Contractor Begins Site Specific Response Spectrum
10-01-77C
Contractor Completes Site-Specific Resp. Spec. Issues Draft Final Rpt.
04-79C
Comments on Draft Final Rpt. (3.0) Rec'd from DOR/SEP
8-79C
Comments on Draft Final Rpt. (3.0) Rec'd from Review Committee
08-79C
Contractor Issues Draft Report (3.0)
08-79C
Review Recommendation Sent to NRR for Approval
11-01-79C
09-79T

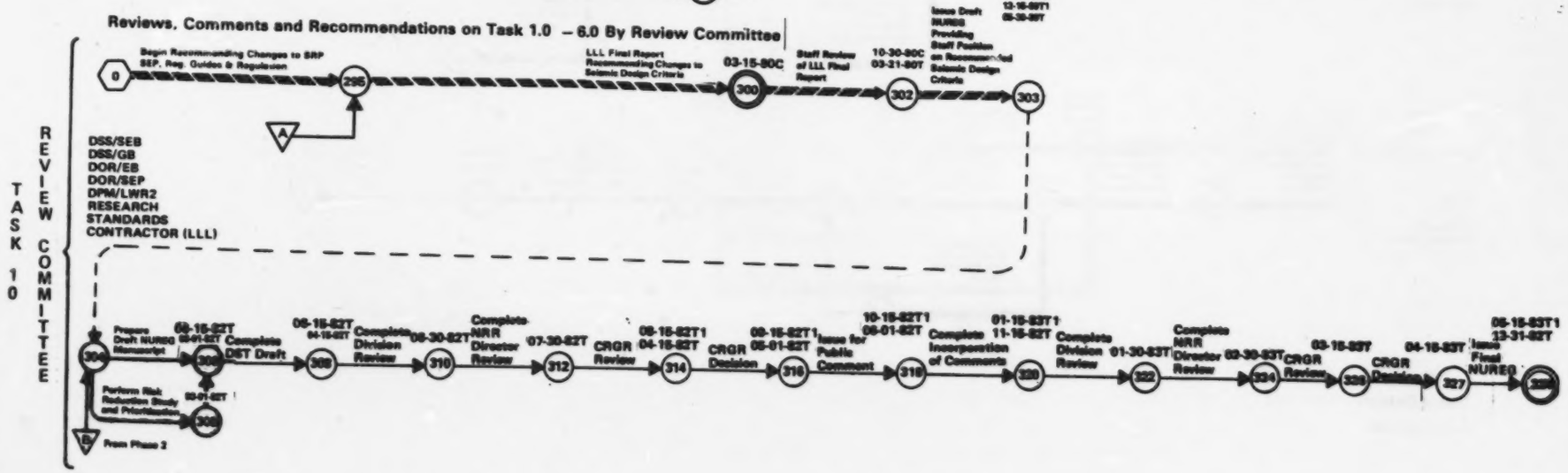
Task 4 Deleted

Seismic After Shocks

SEISMIC DESIGN CRITERIA - SHORT TERM PROGRAM (A-40)

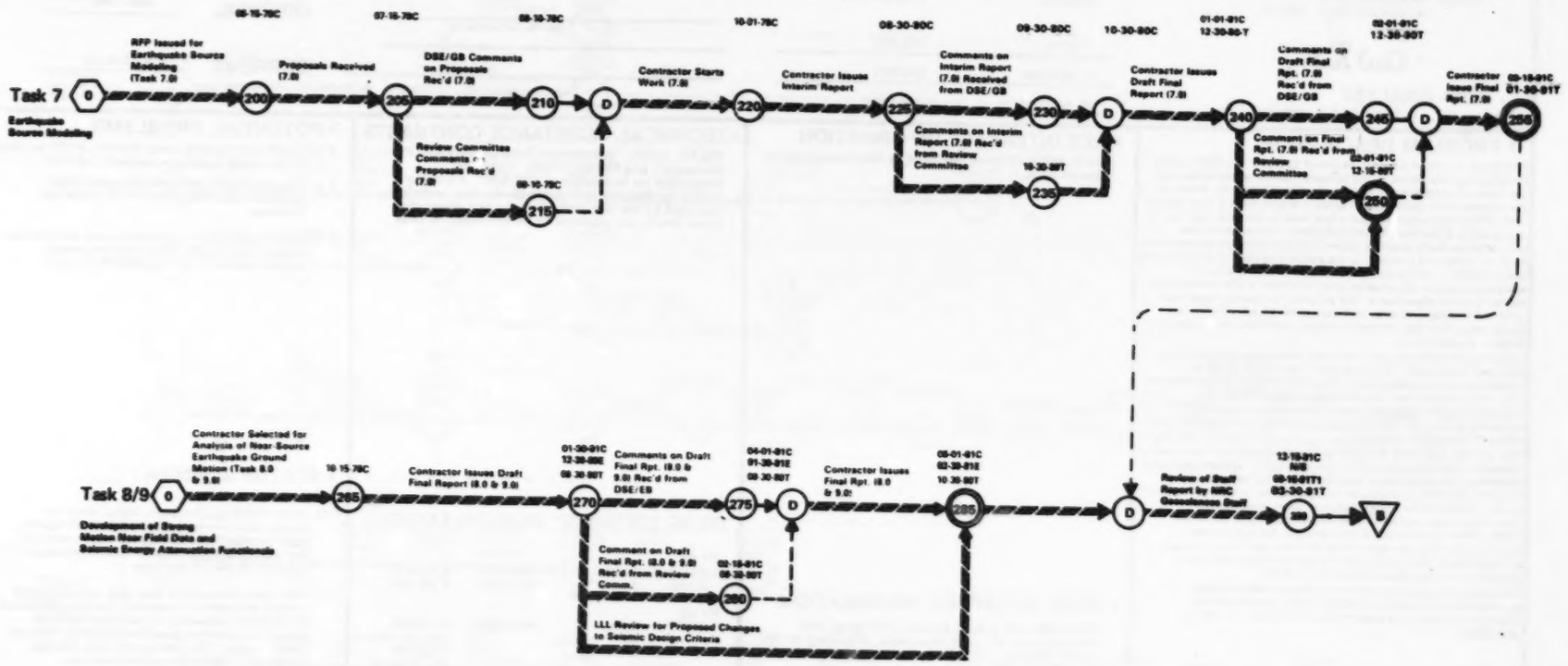
(Phase 1 Cont.)


Task 5
Nonlinear Structural Dynamic Analysis Procedures
Contractor Begins Non-Linear Structural Dynamic Analysis (Task 5.0)
10-01-77C
Contractor Completes Sub-Task 5.1 & Issues Interim Report
12-31-77C
Contractor Completes Sub-Task 5.3 & Completes Draft of Final Report
09-15-78C
Comments on Draft Final Rpt. (5.0) Rec'd from SEB
09-22-78C
Contractor Issues Final Report (5.0)
09-30-78C
Final Rpt. (5.0) & Reviews Recs. Sent to NRR For Approval
11-01-79C
09-30-79T
Comments on Draft Final Rpt. (5.0) Rec'd from Review Committee
09-22-78C
Task 6
Soil Structure Interaction
Contractor Begins Soil-Structure Interaction (Task 6.0)
10-01-77C
Contractor Completes Sub-Task 6.1 & Issues Interim Report
12-31-77C
Contractor Issues Draft on Final Rpt. (6.0)
09-01-78C
Comments on Draft Final Rpt. (6.0) Rec'd From DSS/SEB
09-15-78C
Contractor Issues Final Report (6.0)
02-79C
Final Rpt. (6.0) & Review Recommendation Sent to NRR for Approval
11-01-79C
09-30-79T
Comments on Draft Final Rpt. (6.0) Rec'd from Review Committee
09-15-78C



Phase 2

SEISMIC DESIGN CRITERIA - SHORT TERM PROGRAM (A-40)



CONTAINMENT EMERGENCY SUMP PERFORMANCE (A-43)

AS OF WEEK ENDING MAY 21, 1982

KEY PERSONNEL

TASK MANAGER
A. SERKIZ (DST) X24217

Aw Serkiz

MPA ANALYST
D. SCHWARZ X27106

TASK REVIEWERS

NAME	BRANCH
F. ORR	RSB/DSI
P. NORIAN	GIB/DST
W. BUTLER	CSB/DSI

SCHEDULED COMPLETION

ORIGINAL: April 1982

CURRENT: # 07-15-83

• PROBLEM DESCRIPTION

Following a Loss of Coolant Accident (LOCA) in a PWR, water flowing from the break in the primary system would collect on the floor of containment. During the injection mode, water for core cooling and containment spray is drawn from a large supply tank. When the water reached a low level in the tank, pumps are realigned to draw from the containment. This is called the recirculation mode wherein water is drawn from the containment floor or sump and pumped to the primary system or containment spray headers. This program addresses the safety issue of adequate sump or suppression pool function in the recirculation mode. It is the objective of this program to develop improved criteria for design, testing, and evaluation which will provide better assurance that emergency sumps will function to satisfy system requirements.

The principal concerns are somewhat interrelated but are best discussed separately. One deals with the various kinds of insulation used on piping and components inside of containment. The concern being that break-initiated debris from the insulation could cause blockage of the sump or otherwise adversely affect the operation of the pumps, spray nozzles, and valves of the safety systems.

The second deals with the hydraulic performance of the sump as related to the hydraulic performance to safety systems supplied therefrom. Preoperational tests have been performed on a number of plants to demonstrate operability in the recirculation mode. Adverse flow conditions have been encountered requiring design and procedural modifications to eliminate them. These conditions, air entrainment, cavitation, and vortex formation, are aggravated by blockage. If not avoided or suppressed, they could result in pump failure during the long term cooling phase following a LOCA.

The concerns relative to debris, blockage, and hydraulic performance also apply to boiling water reactors during recirculation from the suppression pools, and will also be addressed.

• RES INTERFACE INFORMATION

None; USI A-43 being managed by the Generic Issues Branch (GIB).

• ACRS INTERFACE INFORMATION

The ACRS staff is kept abreast of findings and developments on an informal basis. A briefing on the current findings and status of resolution was held December 3, 1981.

• TECHNICAL ASSISTANCE CONTRACTS

FIN No. A1237, "Containment Emergency Sump Performance", and FIN No. A1296, "Technical Assistance for Resolution of USI A-43", are being funded by RES and NRR respectively. This work is managed by the GIB Task Manager and these combined efforts are expected to be concluded in FY82.

FIN. NO.	CONTRACTOR	OBLIGATED	EXPENDED
FY 81:			
A1237	Sandia	$225,000	$101,000
A1296	Sandia	$120,000	$ 75,100
FY 82:			
# A1237	Sandia	$209,000*	$ 85,000 thru 1/01/82 $125,000 thru 4/01/82
# A1296	Sandia	$150,000	$ 73,400 thru 1/01/82 $186,000 thru 4/01/82

#* This 1/12/82 RES is less than the $275,000 guidance set forth to Sandia and insufficient to conclude FY 82 efforts. DST funding to make up the difference will be required and plans are to augment A1296 funding accordingly.

• POTENTIAL PROBLEMS

1) Insufficient funding under FIN No. A1237 to conclude FY 82 efforts leading to resolution of A-43.

2) Slippage of final resolution date due to uncertainties associated with CRGR review and feedback.

#3) CRGR-required risk and costs impact analyses have become a pacing item since external, subcontracted efforts will be required.

• STATUS SUMMARY

#1) Alden Research Laboratory (ARL) sump hydraulic tests have been completed. Draft reports, reporting significant findings have been received and these results have been utilized to develop A-43 technical resolution positions.

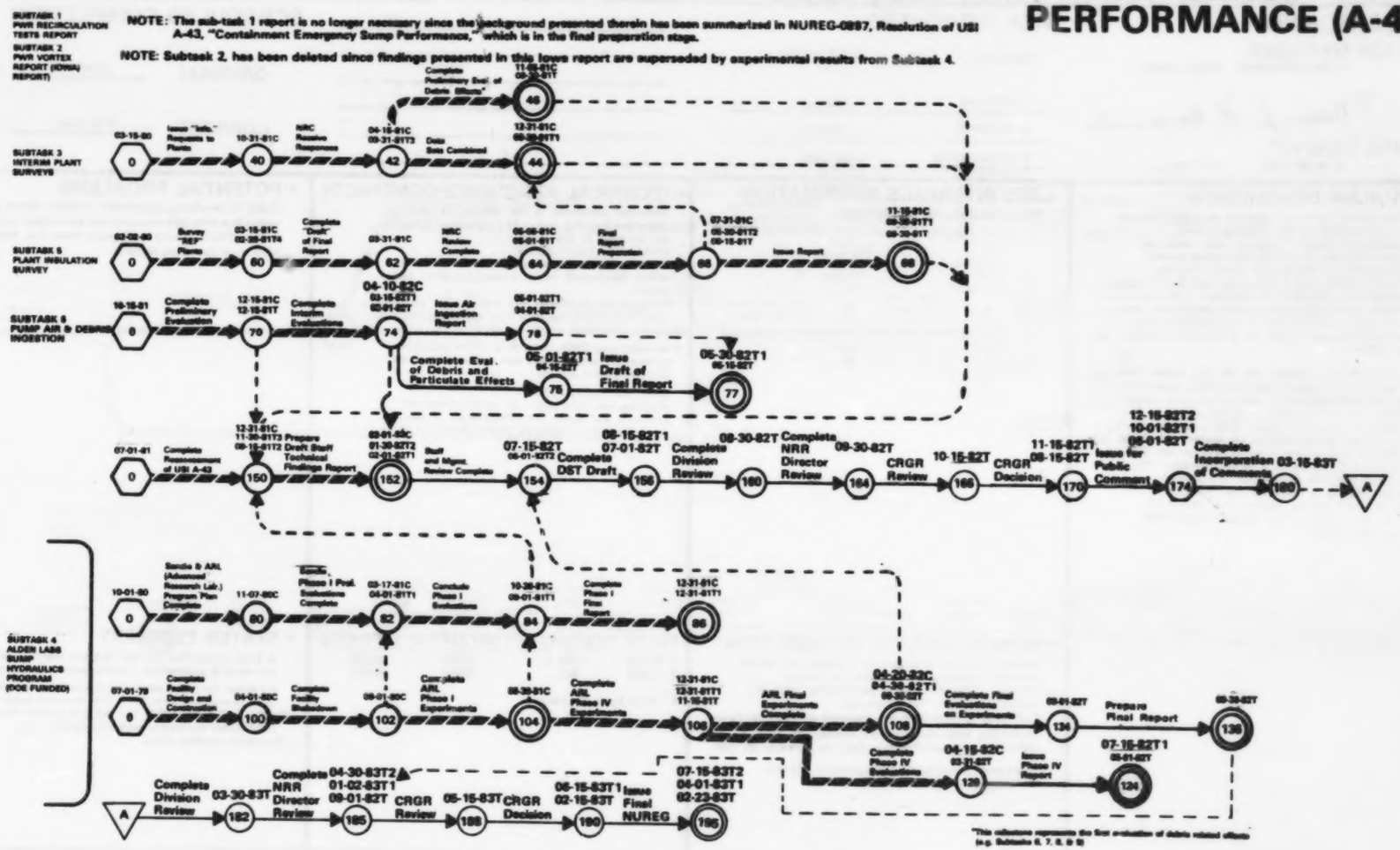
#2) Plant insulation surveys have been completed and a debris calculational method(s) has been developed. These debris evaluation models have been subjected to both internal staff and external technical "peer" reviews. The final report, with five illustrative plant calculations is scheduled for completion in May 1982.

#3) Pump air and particulate ingestion effects have been evaluated and a final report is scheduled for completion in May 1982.

#4) A resolution NUREG (NUREG-0897 Draft) has been prepared, has received staff and contractor review, and is being typed final for initiation of NRR management concurrence in July 1982.

# INDICATES A CHANGE IN INFORMATION

CONTAINMENT EMERGENCY SUMP PERFORMANCE (A-43)


SUBTASK 1 PWR RECIRCULATION TESTS REPORT
NOTE: The sub-task 1 report is no longer necessary since the background presented therein has been summarized in NUREG-0897, Resolution of USI A-43, "Containment Emergency Sump Performance," which is in the final preparation stage.
SUBTASK 2 PWR VORTEX REPORT (IOWA REPORT)
NOTE: Subtask 2, has been deleted since findings presented in this Iowa report are superseded by experimental results from Subtask 4.
SUBTASK 3 INTERIM PLANT SURVEYS
SUBTASK 5 PLANT INSULATION SURVEY
SUBTASK 6 PUMP AIR & DEBRIS INGESTION
Complete Preliminary Eval. of Debris Effects*
Issue Draft of Final Report
Complete Eval. of Debris and Particulate Effects
Prepare Draft Staff Technical Findings Report
Staff and Mgmt. Review Complete
Complete DST Draft
Complete Division Review
Complete NRR Director Review
CRGR Review
CRGR Decision
Issue for Public Comment
Complete Incorporation of Comments
Issue Final NUREG
SUBTASK 4 ALDEN LABS SUMP HYDRAULICS PROGRAM (DOE FUNDED)
Complete ARL Phase I Experiments
Complete ARL Phase IV Experiments
ARL Final Experiments Complete
Complete Final Evaluations on Experiments
Prepare Final Report
Issue Phase IV Report
Complete Division Review
Complete NRR Director Review
CRGR Review
CRGR Decision
Issue Final NUREG

*This milestone represents the final evaluation of debris related effects (e.g. Subtasks 6, 7, 8, & 9)

STATION BLACKOUT (A-44)

AS OF WEEK ENDING MAY 21, 1982

KEY PERSONNEL

TASK MANAGER
P. BARANOWSKY (RES) X28980

P. Noonan for P. Baranowsky

MPA ANALYST
D. SCHWARZ X27106

TASK REVIEWERS

NAME	BRANCH
C. NELSON	ORB/DL
O. CHOPRA	PSB/DSI
F. CLEMENSON	ASB/DSI

SCHEDULED COMPLETION

ORIGINAL JUNE 1982

CURRENT #10-15-83

• PROBLEM DESCRIPTION

Electric power for safety systems at nuclear power plants is supplied by two redundant and independent divisions. Each of these electrical divisions includes an offsite alternating current (A.C.) source, an onsite A.C. source (usually diesel-generators), and a direct current (D.C.) source. Appendix A to 10 CFR 50 defines a total loss of offsite power as an anticipated occurrence, and as such, it is required that an independent emergency onsite power supply be provided at nuclear power plants.

The unlikely, but possible loss of A.C. power (that is, the loss of A.C. power from the offsite source and from the onsite source) is referred to as a station blackout. In the event of a station blackout, the capability to cool the reactor core would be dependent on the availability of systems which do not require A.C. power supplies, and on the ability to restore A.C. power in a timely manner. The concern is that the occurrence of a station blackout may be a relatively high probability event and that the consequences of this event may be unacceptable, for example, severe core damage may result.

• RES INTERFACE INFORMATION

RES is managing the Task for NRR

• ACRS INTERFACE INFORMATION

Station Blackout is related to a number of ACRS concerns regarding the reliability of power systems. This task will be coordinated with the committee as the task progresses.

Task action plan outline was discussed with AC/DC Power Reliability Subcommittee on January 22, 1981. Another subcommittee briefing was held on October 30, 1981.

#A program status briefing was given to this subcommittee on March 30, 1982.

• TECHNICAL ASSISTANCE CONTRACTS

Technical Assistance in the amount of 800K is estimated for this task. The following elements are included in this program:

ORNL FIN B0744 $560K --
Evaluate expected frequency and duration of offsite (preferred) power losses at nuclear power plants.
Estimate the reliability and evaluate the dominant factors affecting the reliability of emergency A.C. power supplies.

#An additional 500K in funding will be added to perform statistical correlation and trend analysis of diesel generator data.

SNL FIN A1302 $290K --
Evaluate the risks posed by station blackout accidents and assess the effectiveness of safety improvements in reducing these risks.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED*
# B0744	ORNL	460K	362K
# A1302	SNL	290K	234K

*As of April 30, 1982

• POTENTIAL PROBLEMS

Delay in receiving approvals to obtain diesel generator reliability data has compressed scheduled work at ORNL. This resulted in slippage from Feb. 1982 for completion of the draft reports.

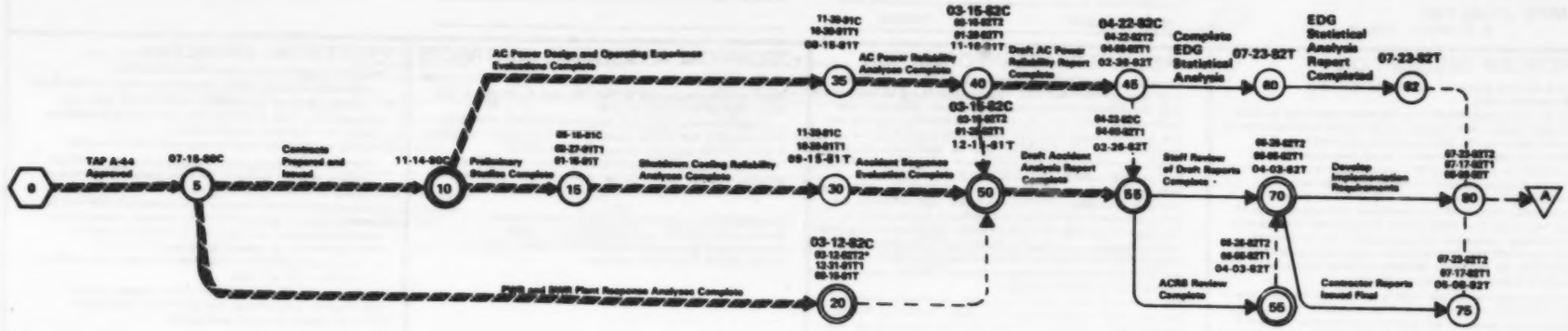
• STATUS SUMMARY

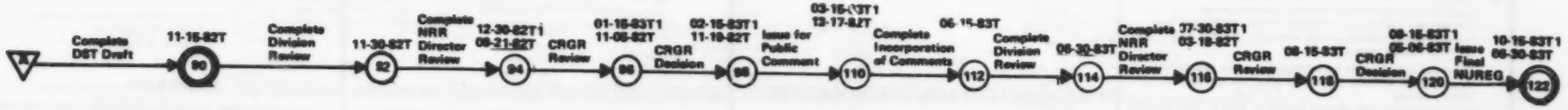
A Task Action Plan for this task was issued in July 1980 and work is proceeding in accordance with this plan.

A request for diesel generator reliability data was sent to all licensees in July 1981. The first data began to #arrive in October 1981. The last data arrived March 1982. Draft contractor technical reports were #provided by May 1982.

# INDICATES A CHANGE IN INFORMATION

STATION BLACKOUT (A-44)





*Preliminary analyses completed 9-81.

SHUTDOWN DECAY HEAT REMOVAL REQUIREMENTS (A-45)

AS OF WEEK ENDING MAY 21, 1982

KEY PERSONNEL

TASK MANAGER
ANDREW MARCHESE X34712

A. D. Marchese

MPA ANALYST
B. ST. MARY X28330

TASK REVIEWERS

NAME	BRANCH
#T. MARSH	RSB/DSI
F. ROSA	ICSB/DSI
M. SRINIVASAN	PSB/DSI
E. McPEEK	SSPB/DL
C. NELSON	ORB-3/DL
#M. CUNNINGHAM	DRA/RES

SCHEDULED COMPLETION

ORIGINAL	#10/85 PRELIMINARY NETWORK
CURRENT	#09-30-85

• PROBLEM DESCRIPTION

#Task A-45 was approved as a USI by the NRC in December 1980.

Although many improvements to the steam generator auxiliary feedwater system were required of the reactor manufacturers by the NRC following the TMI-2 accident, the staff feels that providing an alternative means of decay heat removal could substantially increase the plants' capability to deal with a broader spectrum of transients and accidents and potentially could, therefore, significantly reduce the overall risk to the public. Consequently, Task A-45 will investigate alternative means of decay heat removal in PWR plants, including but not limited to using existing equipment where possible. This Unresolved Safety Issue will also investigate the need and possible design requirements for improving reliability of decay heat removal systems in boiling water reactors (BWRs).

The overall purpose of Task A-45 is to evaluate the adequacy of current licensing design requirements, in order to ensure that nuclear power plants do not pose an unacceptable risk due to failure to remove shut-down decay heat. The objective will be to develop a comprehensive and consistent set of shutdown cooling requirements for existing and future LWRs, including the study of alternative means of shutdown decay heat removal and of diverse "dedicated" systems for this purpose.

The approach taken to this problem comprises the following subtasks:

#– Development of criteria to judge acceptability of decay heat removal systems in existing & future plants.
– Development of means for improvement of DHR systems.
– Assessment of existing plants to identify those in which DHR systems require improvement.
– Development of a plan for implementing new requirements, if any, for DHR systems required to meet the acceptance criteria above.

The interrelation and relative timing of each of the sub-tasks is shown on the schedule network.

• RES INTERFACE INFORMATION

Close coordination and cooperation will be required on Task A-45 between NRR and RES. RES assistance will be required from the Divisions of Risk Analysis, Accident Evaluation, and Engineering Technology. The Division of Risk Analysis will provide technical input from their Sandia Laboratory Program on Alternate Decay Heat Removal Concepts, technical evaluations relative to reliability and risk assessment for shutdown decay heat removal systems, and input from Task A-44, "Station Blackout," relative to shutdown cooling systems. The Division of Accident Evaluation will provide technical input relative to the transient response of existing and improved shutdown decay heat removal systems to transient events and small LOCAs. This will also include performing (in-house, contractors) detailed thermal-hydraulics analyses where required to support existing and improved decay heat removal systems behavior under transient and accident conditions.

The Division of Engineering Technology will provide assistance in the preparation and publication (i.e., Rules, Regulatory Guides, Standard Review Plans) of a consistent and comprehensive set of shutdown decay heat removal requirements. At the outset, particular attention will need to be focused on the further development of Regulatory Guide 1.139, "Guidance for Residual Heat Removal to Achieve and Maintain Cold Shutdown," Revision 1.

• ACRS INTERFACE INFORMATION

– Task Manager briefed the full committee on Sept. 10, 1981 #on the present plans and content of the A-45 Task Action Plan (TAP).
#– ACRS Subcommittee on Decay Heat Removal Systems (DHRS) met on March 16, 1982, and Task Manager provided the subcommittee with an update of the TAP.
– Further meetings with the full committee and subcommittee on DHRS will be held as the work on A-45 progresses and certain pre-determined milestones are completed.

• TECHNICAL ASSISTANCE CONTRACTS

– Supplemented UCLA Contract (FIN-B8835) by 10K to provide direct technical assistance to Task Manager to develop a Task Action Plan for resolution of A-45.
– Implemented a technical assistance contract with BNL (FIN-A-3381) for $100K to make a determination whether it is feasible to separate all currently operating U.S. commercial LWR's into groups based on whether #plants covered in WASH-1400, RSSMAP, and IREP. they have similar DHRS design characteristics to the six

#Implemented a technical assistance contract with Sandia (FIN A1309) to provide overall project management, technical direction and integration for the entire Task A-45 program.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
B8835	UCLA	$10,000	$10,000
# A3381	BNL	$100,000	$74,000
#A1309	Sandia	$400,000	–

• POTENTIAL PROBLEMS

Potential problem areas identified to date are:

– Obtaining sufficient NRR manpower to work on Task A-45.
#– Development of appropriate reliability or quantitative goals for Task A-45 and translation of probabilistic results into licensing requirements.
#– Program funding must be approved and obtained.
– Obtaining necessary operating experience on shutdown cooling systems including the most current DHR system design information resulting from Post-TMI changes.
– Uncertainty in the quality of information that will be #available from ongoing reliability and risk assessments, on what schedule, and whether information can be extrapolated to all operating plants.
– Close coordination and cooperation needed within NRR and between NRR and RES.
– Number of plants that need to be assessed may be significantly greater than the plants that will have a #PRA or reliability study performed.
#– Time to obtain approval of funding by NRR and the Senior Contract Review Board has been greater than expected by about six months (See Status Summary).

• STATUS SUMMARY

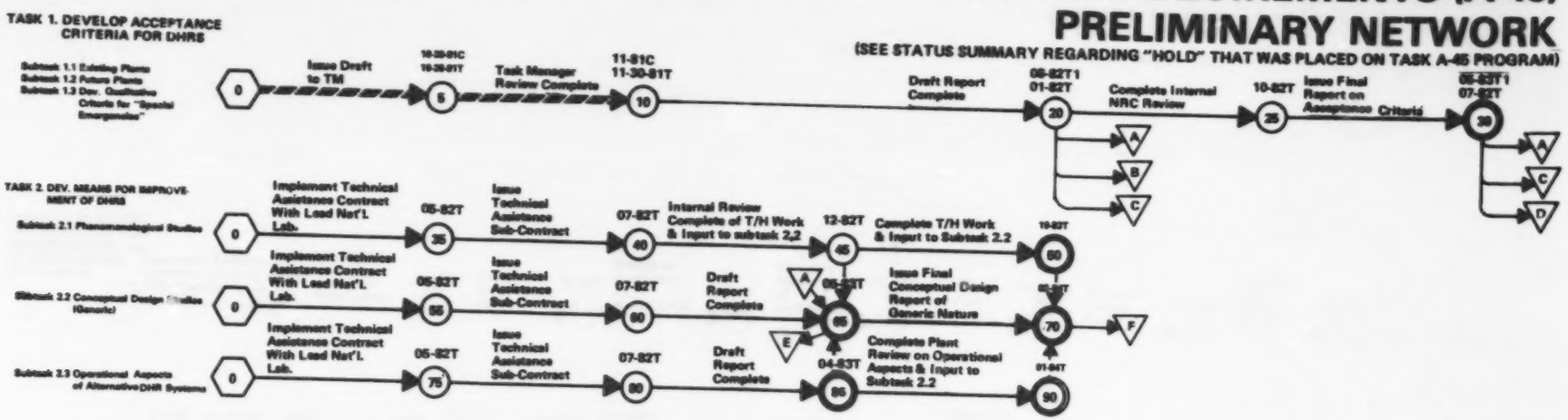
The task manager has been selected and a draft Task Action Plan was published on May 22, 1981.

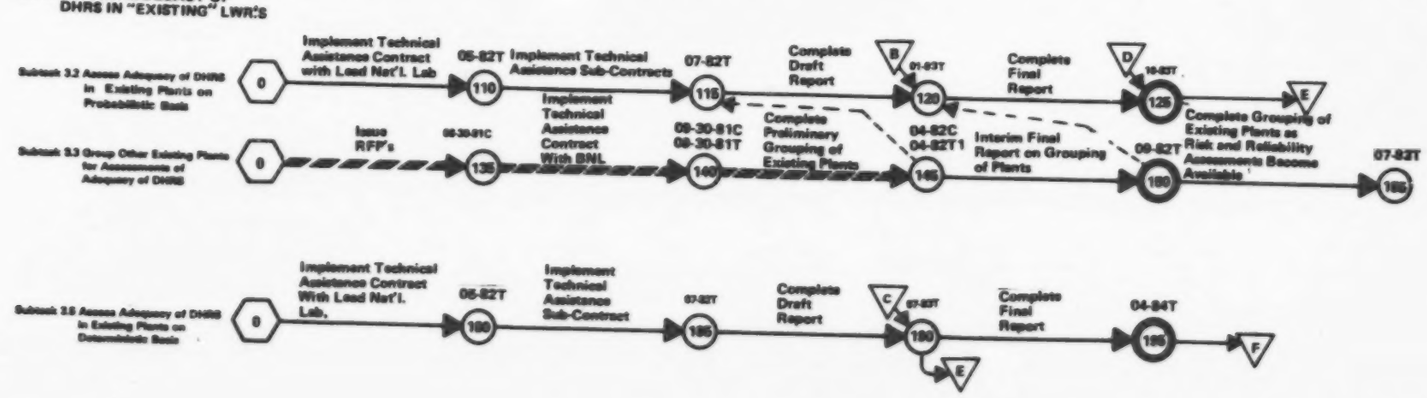
Based on internal staff and ACRS comments, a revised TAP was approved on October 7, 1981.

#The Director of NRR did not approve the requested funding level for Task A-45 technical assistance work, and a "HOLD" was placed on the program in Nov. 1981. Various options were considered for Task A-45, most of which included reduced scope, funding and completion time. A recommended option was approved by The Director of NRR in Feb. 1982. A technical assistance contract package was approved by the Senior Contract Review Board on April 9, 1982 and implemented with the Contractor (Sandia) on May 10, 1982.

SHUTDOWN DECAY HEAT REMOVAL REQUIREMENTS (A-45)
PRELIMINARY NETWORK

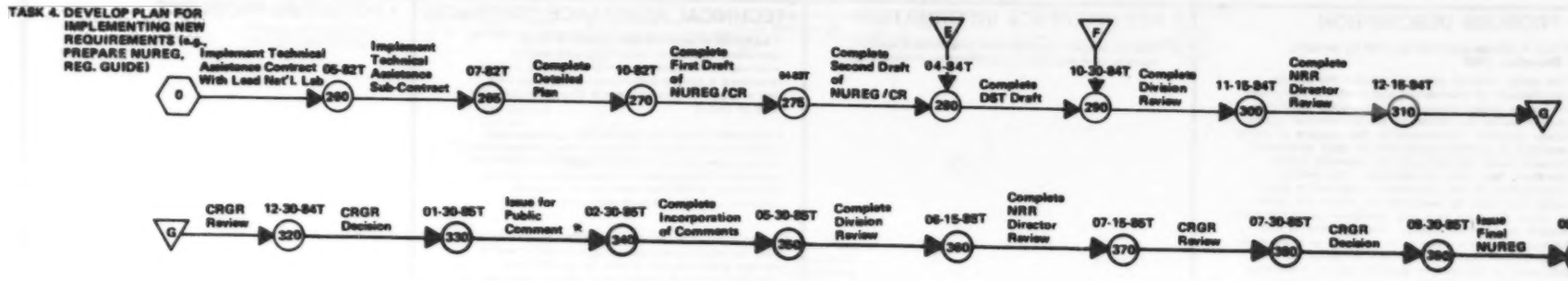
(SEE STATUS SUMMARY REGARDING "HOLD" THAT WAS PLACED ON TASK A-45 PROGRAM)





SHUTDOWN DECAY HEAT REMOVAL REQUIREMENTS (A-45) (CONTINUED)
PRELIMINARY NETWORK

(SEE STATUS SUMMARY REGARDING "HOLD" THAT WAS PLACED ON TASK A-45 PROGRAM.)



SEISMIC QUALIFICATION OF EQUIPMENT IN OPERATING PLANTS (A-46)

AS OF WEEK ENDING MAY 21, 1982

KEY PERSONNEL

TASK MANAGER
PATRICK SEARS X24216
Patrick Sears

MPA ANALYST
B. ST. MARY X28330

TASK REVIEWERS

NAME	BRANCH
JERRY JACKSON	EQB/DE

SCHEDULED COMPLETION

ORIGINAL 12-15-83

CURRENT #05-01-85

• PROBLEM DESCRIPTION

#Task A-46 was approved as a USI by the NRC December, 1980.

The design criteria and methods for the seismic qualification of mechanical and electrical equipment in nuclear power plants have undergone significant change during the course of the commercial nuclear power program. Consequently, the margins of safety provided in existing equipment to resist seismically induced loads and perform the intended safety functions may vary considerably. The seismic qualification of the equipment in operating plants must, therefore, be reassessed to ensure the ability to bring the plant to a safe shutdown condition when subject to a seismic event. The objective of this Unresolved Safety Issue is to establish an explicit set of guidelines that could be used to judge the adequacy of the seismic qualification of mechanical and electrical equipment at all operating plants in lieu of attempting to backfit current design criteria for new plants. This guidance will concern equipment required to safely shutdown the plant, as well as equipment whose function is not required for safe shutdown, but whose failure could result in adverse conditions which might impair shutdown functions. Also, explicit guidelines will be established for use in requalifying equipment whose seismic qualification was found to be inadequate.

A preliminary breakdown of the tasks is as follows:

#	Task 1	Develop Basis for Equipment to be Qualified
#	Task 2	Survey of Equipment Qualification Methods
#	Task 3	Develop Methods for In-Situ Qualification of Equipment in Operating Plants
#	Task 4	Seismic Qualification of Equipment Using Non-Nuclear Plant Dynamic Response Information
#	Task 5	Develop Guidelines for Qualification Response Spectra
	Task 6	Document Results of Task A-46 and Prepare a Final NUREG

• RES INTERFACE INFORMATION

RES has placed a contract with Southwest Research Institute to survey seismic qualification methodology.

• ACRS INTERFACE INFORMATION

A preliminary Task Action Plan was presented to the ACRS. The staff plans to meet with the ACRS at appropriate times to present the status of program results.

• TECHNICAL ASSISTANCE CONTRACTS

A statement of work has been written to study appropriate methods of requalifying equipment in operating plants. Laboratory and in-situ testing are being considered. A funding authorization of $60K has been sent to Edgerton, Germeshausen & Grier (EG&G) to start work on Task 3.

A statement of work has been written to review and analyze inservice dynamic response data observed in non-nuclear power plants and industrial facilities to aid in the qualification of equipment in nuclear plants. A funding authorization of $80K has been sent to Lawrence Livermore Laboratory (LLL) to start work on Task 2.

#A Statement of Work has been written to develop a basis for selecting equipment to be qualified and to develop guidelines and criteria for qualification response spectra. A funding of $126K has been sent to Brookhaven National Laboratory (BNL) for this work. This contract is under EQB cognizance.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
#A0423	LLNL	$50K	$10K (est.)
#A6474	INEL	$50K	$10K (est.)
#A3397	BNL	$126K	$20K (est.)

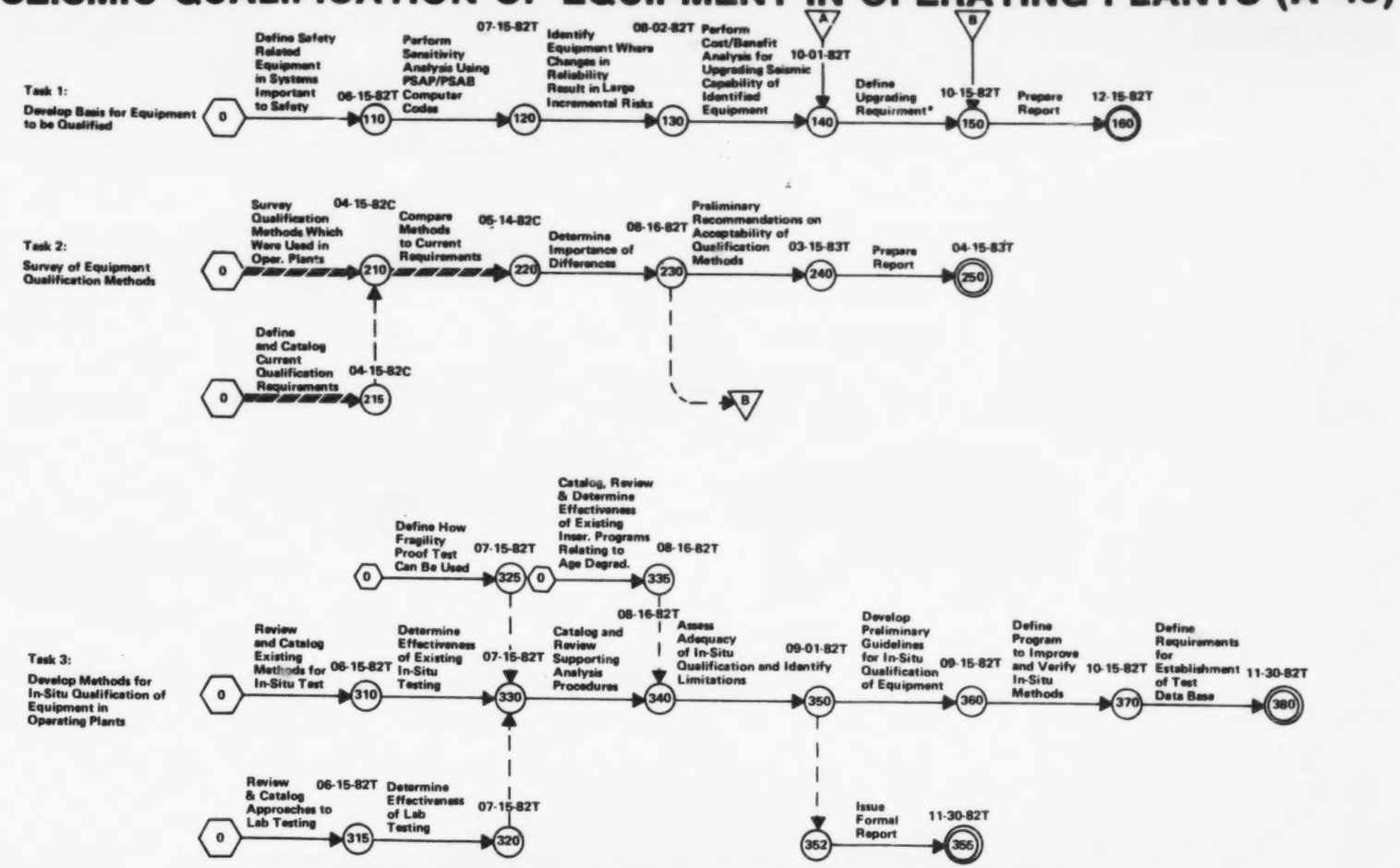
• POTENTIAL PROBLEMS

• STATUS SUMMARY

#Work has been initiated on all tasks with all contractors well into their first subtasks. A walkthrough of non-nuclear power plants which survived the 1971 San Fernando Earthquake has been conducted. Meetings have taken place with members of the Owner's Group who are surveying nuclear and non-nuclear equipment.

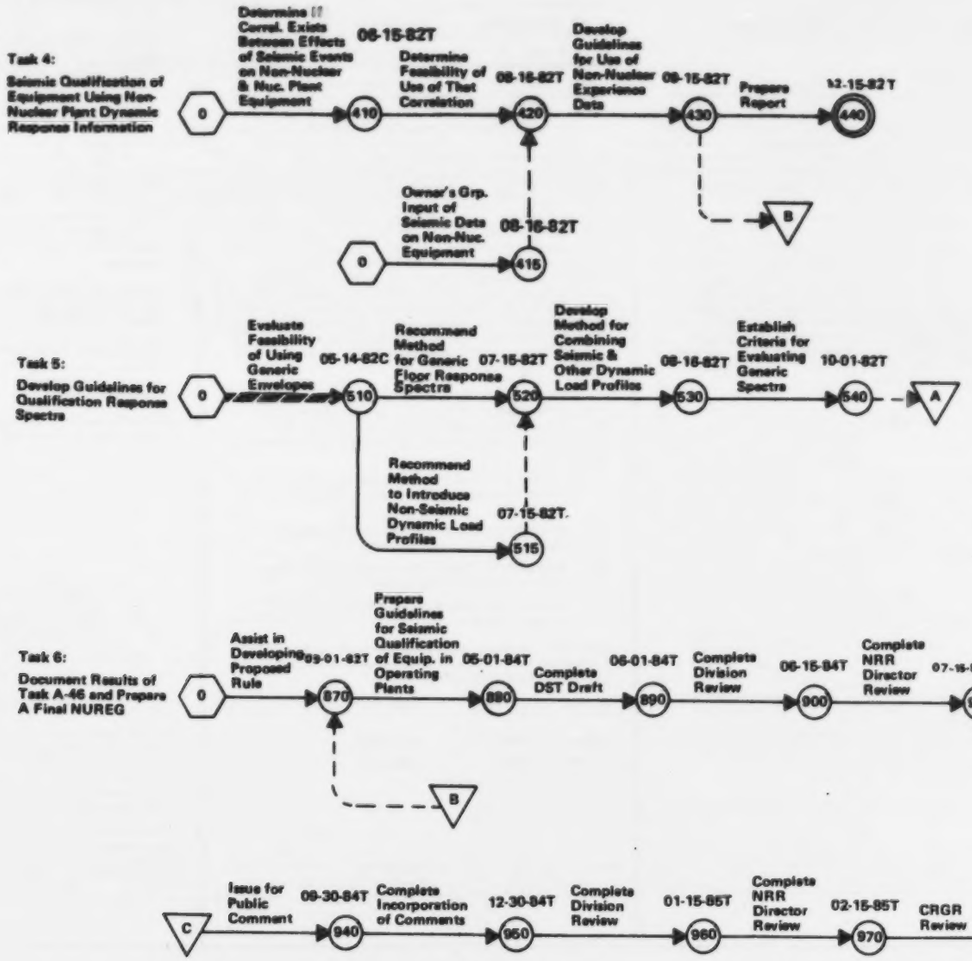
3R # INDICATES A CHANGE IN INFORMATION

SEISMIC QUALIFICATION OF EQUIPMENT IN OPERATING PLANTS (A-46)



*Required Upgrading: (1) Upgrade None, (2) Upgrade High Risk Only, (3) Upgrade All.

SEISMIC QUALIFICATION OF EQUIPMENT IN OPERATING PLANTS (A-46) Continued

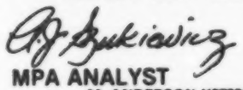


SAFETY IMPLICATIONS OF CONTROL SYSTEMS (A-47)

AS OF WEEK ENDING MAY 21, 1982

KEY PERSONNEL

TASK MANAGER
ANDREW SZUKIEWICZ (GIB) X24713



MPA ANALYST
M. ANDERSON X27021

TASK REVIEWERS

NAME	BRANCH
D. BASDEKAS	DFO/RES
C. ROSSI	ICSB/DSI
#S. NEWBERRY	RRAB/DST
F. ORR	RSB/DSI
M. CHIRAMAL	PSU/AEOD
J. T. BEARD	ORAB/DL
#W. KENNEDY	PTRB/DHFS

SCHEDULED COMPLETION

1978 ANNUAL REPORT: NOT DETERMINED

CURRENT: N/S

• PROBLEM DESCRIPTION

#Task A-47 was approved as a USI by the NRC in December 1980.

This issue concerns the potential for accidents or transients being made more severe as a result of control system failures or malfunctions. These failures or malfunctions may occur independently or as a result of the accident or transient under consideration and would be in addition to any control system failure that may have initiated the event. Although it is generally believed that control system failures are not likely to result in loss of safety functions which could lead to serious events or result in conditions that safety systems are not able to cope with, in-depth studies have not been performed to support this belief. The potential for an accident that would affect a particular control system - and the effects of the control system failures - will differ from plant to plant. Therefore, it is not likely that it will be possible to develop generic answers to these concerns, but rather plant-specific reviews will be required. The purpose of this Unresolved Safety Issue is to define generic criteria that may be used for plant-specific reviews. A specific subtask of this issue will be to study the steam generator overfill transient in PWRs and the reactor overfill transient in BWRs to determine and define the need for preventive and/or mitigating design measures to accommodate this transient.

• RES INTERFACE INFORMATION

Close coordination will be required on Task A-47 between NRR and RES. RES assistance will be required from the Division of Facilities Operations and the Division of Risk Analysis. The Division of Facilities Operations will provide input from their research programs (currently being developed in conjunction with A-47) on control system interactions. The Division of Risk Analysis will provide input on their current ongoing ORNL program ("Evaluation of Pressurized Thermal Shock"). Other research activities are being evaluated to determine their significance to A-47 and may be identified later.

• ACRS INTERFACE INFORMATION

The ACRS Subcommittee on Electrical Power Systems (Dr. Kerr) met on the issue on 1/23/81 to initiate an approximate six month study at the request of Chairman Ahearne.

Elements of conceptual Task Action Plan were discussed with ACRS on 06-07-81.

• TECHNICAL ASSISTANCE CONTRACTS

During the week of July 20, 1981, meetings were held #with INEL, EG&G (Idaho Falls), Battelle Northwest and Lawrence Livermore Labs. for the purpose of discussing the A-47 objectives and to assess the type of activities that could best be conducted by these organizations. Similar meetings with ORNL and Sandia Labs. #were held July 9, 1981. A Technical Assistance (T.A.) contract with ORNL (FIN B-0467) was established through NRR/RES. RES will conduct a review of two different types of PWR designs, and perform the activities identified in Tasks 1, 2, 3 and 4 of Task Action Plan A-47. A separate T.A. contract to perform the review on one BWR type design was established with INEL (FIN A-6477). A T.A. contract to review a third PWR design will be established in late FY1982. The technical assistance contracts will perform the following tasks:

1. Evaluate Control System Failures Leading to Steam Generator and/or Reactor Overfill Transients
2. Evaluate Control System Failures Leading to Reactor Overcooling Transients
3. Evaluate (All Others) Control System Actions That Have Safety Implications
4. Evaluate the Effects of Loss of Power to the Control Systems

Additional Technical Assistance Contract work will be needed for Task A-47 and is being defined as part of the developing Task Action Plan.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
# A-6477	INEL	$ 50,000	
# B-0467	ORNL	$600,000	

• POTENTIAL PROBLEMS

Obtain necessary operating experience information on Non-Safety Control Systems for the Nuclear Steam System (NSS) and the Balance of Plant (BOP) systems.

Obtain necessary design information (i.e., schematics and systems descriptions) for the Non-Safety Control System designs for the NSS and the BOP designs.

Traditionally, licensees were not required to provide design or operating experience on Non-Safety Systems and therefore this information may be difficult to obtain.

#Obtain timely concurrence and approval of Task Action Plan A-47 by ICSB/DSI/NRR.

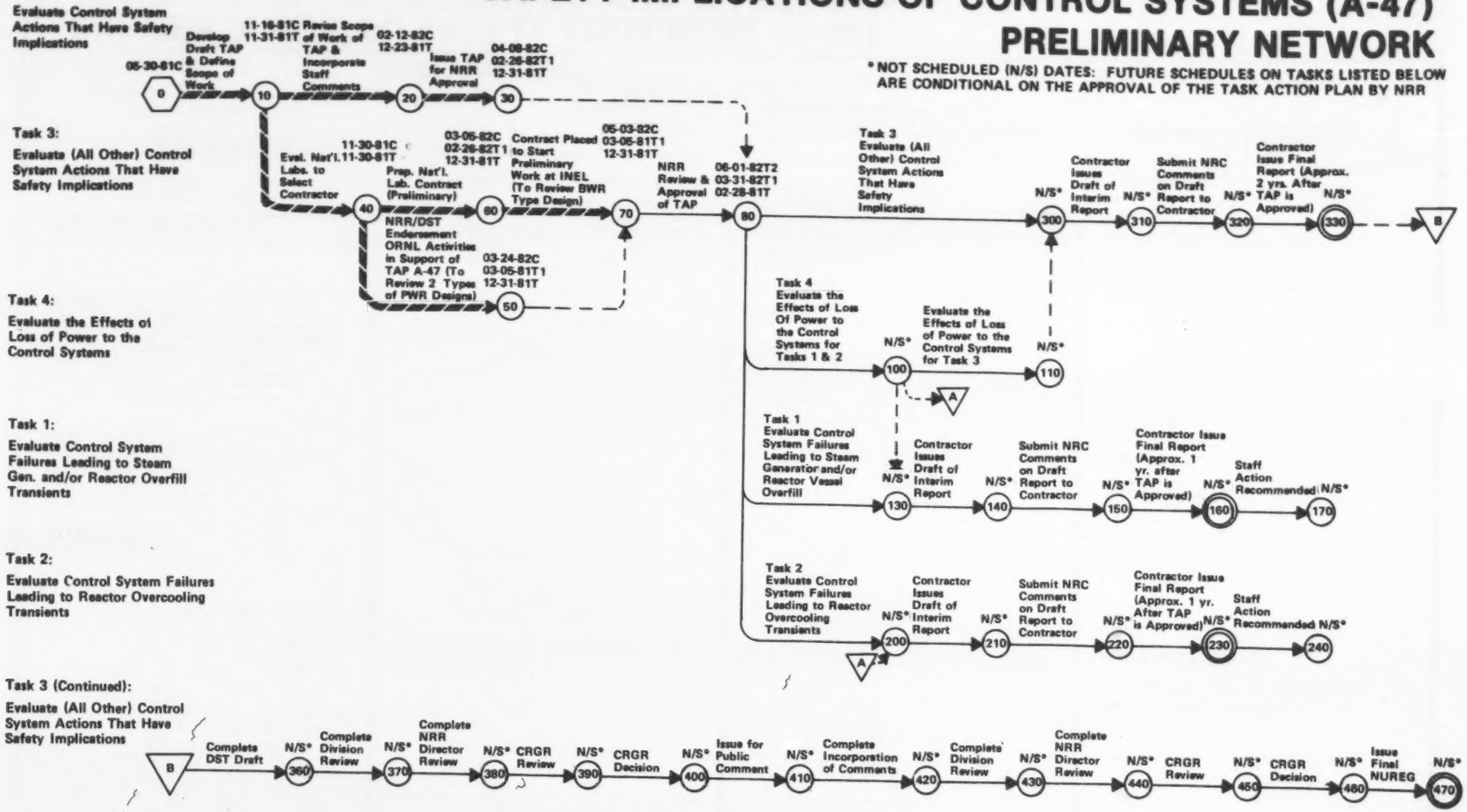
• STATUS SUMMARY

#A task manager has been selected. A revised Draft Task Action Plan was issued in April, 1982 for staff concurrence and approval. A conceptual network has been developed and the task work schedules are being developed. Management approval of the Task Action Plan is expected by 06/01/82.

#A work order was placed with INEL (May 3, 1982) to start preliminary work. Model development for computer simulation of the control systems is underway at ORNL.

SAFETY IMPLICATIONS OF CONTROL SYSTEMS (A-47)
PRELIMINARY NETWORK

*NOT SCHEDULED (N/S) DATES: FUTURE SCHEDULES ON TASKS LISTED BELOW ARE CONDITIONAL ON THE APPROVAL OF THE TASK ACTION PLAN BY NRR


Evaluate Control System Actions That Have Safety Implications
05-30-81C
Develop Draft TAP & Define Scope of Work
11-16-81C Revise Scope 11-31-81T of Work of TAP & Incorporate Staff Comments
02-12-82C 12-23-81T
Issue TAP for NRR Approval
04-08-82C 02-26-82T1 12-31-81T
0
10
20
30
Task 3: Evaluate (All Other) Control System Actions That Have Safety Implications
11-30-81C 11-30-81T
Eval. Nat'l. Labs. to Select Contractor
03-06-82C 02-26-82T1 12-31-81T
Prep. Nat'l. Lab. Contract (Preliminary)
Contract Placed to Start Preliminary Work at INEL (To Review BWR Type Design)
05-03-82C 03-06-81T1 12-31-81T
40
60
70
NRR Review & Approval of TAP
06-01-82T2 03-31-82T1 02-28-81T
80
Task 3 Evaluate (All Other) Control System Actions That Have Safety Implications
N/S*
300
Contractor Issues Draft of Interim Report
310
Submit NRC Comments on Draft Report to Contractor
320
Contractor Issue Final Report (Approx. 2 yrs. After TAP is Approved)
330
B
NRR/DST Endorsement ORNL Activities in Support of TAP A-47 (To Review 2 Types of PWR Designs)
03-24-82C 03-06-81T1 12-31-81T
50
Task 4: Evaluate the Effects of Loss of Power to the Control Systems
Task 4 Evaluate the Effects of Loss Of Power to the Control Systems for Tasks 1 & 2
Evaluate the Effects of Loss of Power to the Control Systems for Task 3
100
110
A
Task 1: Evaluate Control System Failures Leading to Steam Gen. and/or Reactor Overfill Transients
Task 1 Evaluate Control System Failures Leading to Steam Generator and/or Reactor Vessel Overfill
Contractor Issues Draft of Interim Report
Submit NRC Comments on Draft Report to Contractor
Contractor Issue Final Report (Approx. 1 yr. after TAP is Approved)
Staff Action Recommended
130
140
150
160
170
Task 2: Evaluate Control System Failures Leading to Reactor Overcooling Transients
Task 2 Evaluate Control System Failures Leading to Reactor Overcooling Transients
Contractor Issues Draft of Interim Report
Submit NRC Comments on Draft Report to Contractor
Contractor Issue Final Report (Approx. 1 yr. After TAP is Approved)
Staff Action Recommended
200
210
220
230
240
A
Task 3 (Continued): Evaluate (All Other) Control System Actions That Have Safety Implications
B
Complete DST Draft
Complete Division Review
Complete NRR Director Review
CRGR Review
CRGR Decision
Issue for Public Comment
Complete Incorporation of Comments
Complete Division Review
Complete NRR Director Review
CRGR Review
CRGR Decision
Issue Final NUREG
360
370
380
390
400
410
420
430
440
450
460
470

HYDROGEN CONTROL MEASURES AND EFFECTS OF HYDROGEN BURNS ON SAFETY EQUIPMENT (A-48)

AS OF WEEK ENDING MAY 21, 1982

KEY PERSONNEL

TASK MANAGER
CLIFFORD ANDERSON X27488

Clifford J. Anderson

MPA ANALYST
D. SCHWARZ X27106

TASK REVIEWERS

NAME	BRANCH
NOT ASSIGNED	

SCHEDULED COMPLETION

ORIGINAL: NOT DETERMINED

CURRENT: N/S

• PROBLEM DESCRIPTION

#Task A-48 was approved as a USI by the NRC in December 1980.

Postulated reactor accidents which result in a degraded or melted core can result in generation and release to the containment of large quantities of hydrogen. The hydrogen is formed from the reaction of the zirconium fuel cladding with steam at high temperatures and/or by radiolysis of water. Experience gained from the TMI-2 accident indicates that we may want to require more specific design provisions for handling larger hydrogen releases than currently required by the regulations particularly for smaller, low pressure containment designs.

This USI will investigate means to predict the quantity and release rate of hydrogen following degraded core accidents and various means to cope with large releases to the containment such as inerting of the containment or controlled burning. The potential effects of proposed hydrogen control measures on safety including the effects of hydrogen burns on safety related equipment will be investigated.

• RES INTERFACE INFORMATION

TO BE DEVELOPED

• ACRS INTERFACE INFORMATION

TO BE DEVELOPED

• TECHNICAL ASSISTANCE CONTRACTS

TO BE DEVELOPED

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED

• POTENTIAL PROBLEMS

• STATUS SUMMARY

#A draft Task Action Plan (TAP) was issued for NRC comments on 4/02/82. Work has been suspended pending management decision on scope of TAP.

# INDICATES A CHANGE IN INFORMATION

PRESSURIZED THERMAL SHOCK (A-49)

AS OF WEEK ENDING MAY 21, 1982

KEY PERSONNEL

TASK MANAGER
ROY WOODS X24714

Roy Woods

MPA ANALYST
DIANE SCHWARZ X27106

TASK REVIEWERS

NAME	BRANCH
E. THROM	RSB/NRR
C. JOHNSON	RES
L. LOIS	CPB/DSI
J. STROSNIDER	RES
M. VAGINS	RES
R. JOHNSON	GIB/DST
R. KLECKER	MTEB/DE
N. RANDALL	MTEB/DE
G. VISSING	ORB-4/DL
J. CLIFFORD	PTRB/HFS

SCHEDULED COMPLETION

ORIGINAL	Not Determined
CURRENT	#06-30-83

• PROBLEM DESCRIPTION

This task was designated a USI by the NRC in December 1981.

Neutron irradiation of reactor pressure vessel weld and plate materials decreases the fracture toughness of the materials. The fracture toughness sensitivity to radiation induced change is increased by presence of certain materials such as copper. Decreased fracture toughness makes it more likely that, if a severe overcooling event occurs followed by or concurrent with high vessel pressure, and if a small crack is present on the vessels inner surface, that crack could grow to a size that might threaten vessel integrity.

Severe pressurized overcooling events are improbable since they require multiple failures and improper operator performance. However, certain precursor events have happened that could have potentially threatened vessel integrity if additional failures had occurred and/or if the vessel had been more highly irradiated. Therefore, the possibility of vessel failure due to a severe pressurized overcooling event cannot be ruled out.

• RES INTERFACE INFORMATION

A major portion of the work is being performed under a contract with Oak Ridge National Laboratory through the Division of Risk Analysis, RES (FIN # B0468).

Other major contributors are:

Primary System Integrity Research Program through the Division of Engineering Technology, RES and

Code Applications Program through the Division of Accident Evaluation, RES

• ACRS INTERFACE INFORMATION

Meetings have been held and will be scheduled as necessary with the Subcommittee on Metallic Components and with the full ACRS.

• TECHNICAL ASSISTANCE CONTRACTS

#Contract (B-2510) issued to PNL. PNL will provide their recommendations regarding the necessity of corrective actions, and will provide peer review of recommendations being considered by the NRC staff.

#The following RES contracts are providing technical assistance to the PTS program. These are in addition to the technical assistance contracts which were initiated to specifically address the PTS issue and listed in the table below.

FIN No.	Description
#B0119	HSST
#B8133	LWR Pressure Boundary Integrity
#B5988	Surveillance Dosimetry
#B0415	Pressure Vessel Simulation
#B6224	Dosimetry Meas. Data Base
#B7026	JR Curve
#A3215	Code Assessment and Application
#A6047	Code Assessment and Applications
#A7027	Analytical Res. in LWR Safety
#A7217	TRAC Calc Assistance
#B0468	Pressurized Thermal Shock

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
#B-2510	PNL	$400K	$200K-est.
#A-7272	LANL	$210K-FY82 $210K-FY83	$ 80K-est.
#A-3387	BNL	$210K	$ 94K-est.

• POTENTIAL PROBLEMS

None.

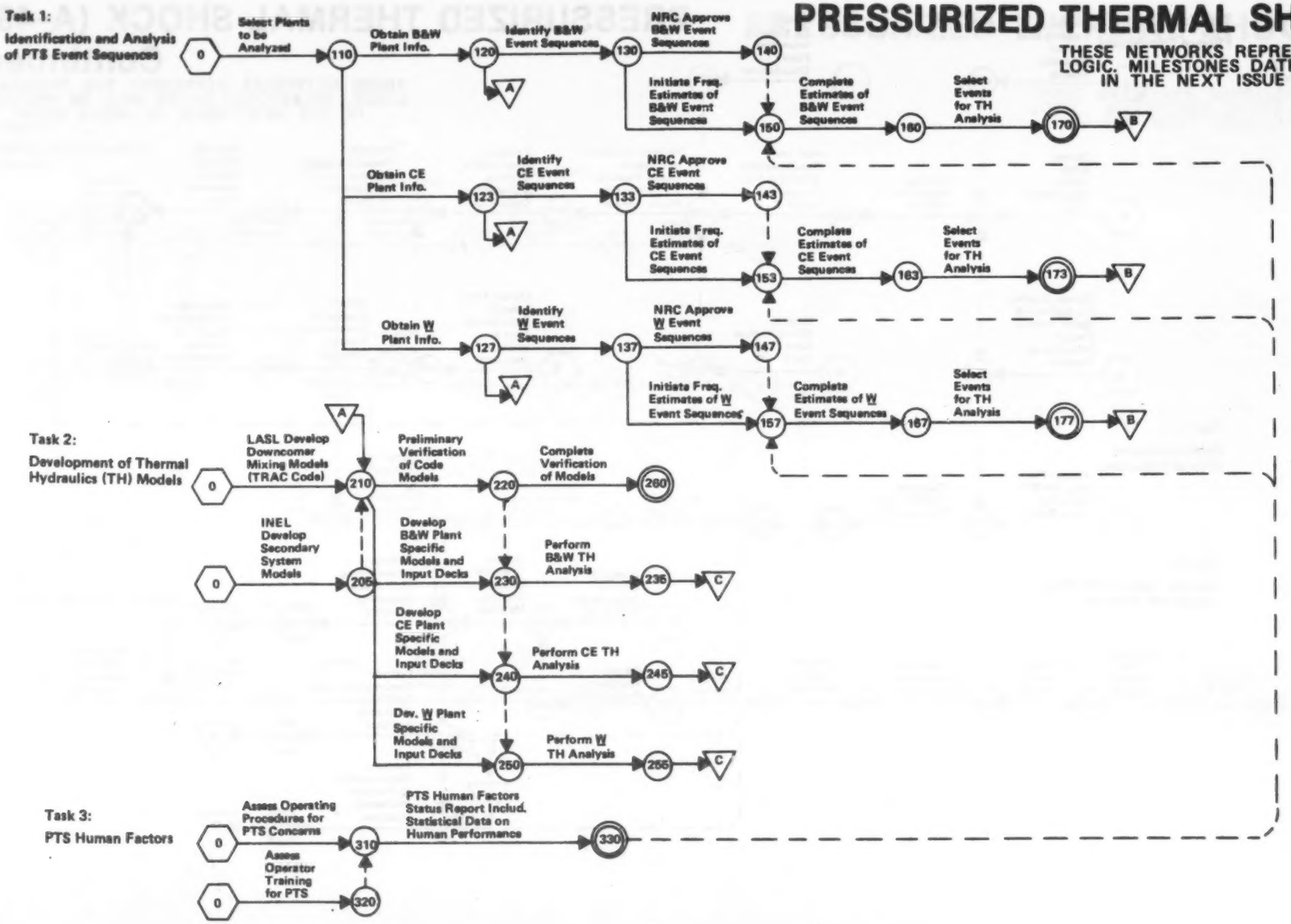
• STATUS SUMMARY

#Draft short-term position is under preparation. Draft PNL proposed position has been received and is under review. ORNL has proposed the first three event sequences to be analyzed and they are under staff review. Review of the "150-day responses" has been completed. Results of the review are being integrated into the June staff position.

#The final Task Action Plan for A-49 was approved and issued on March 26, 1982.

PRESSURIZED THERMAL SHOCK (A-49)

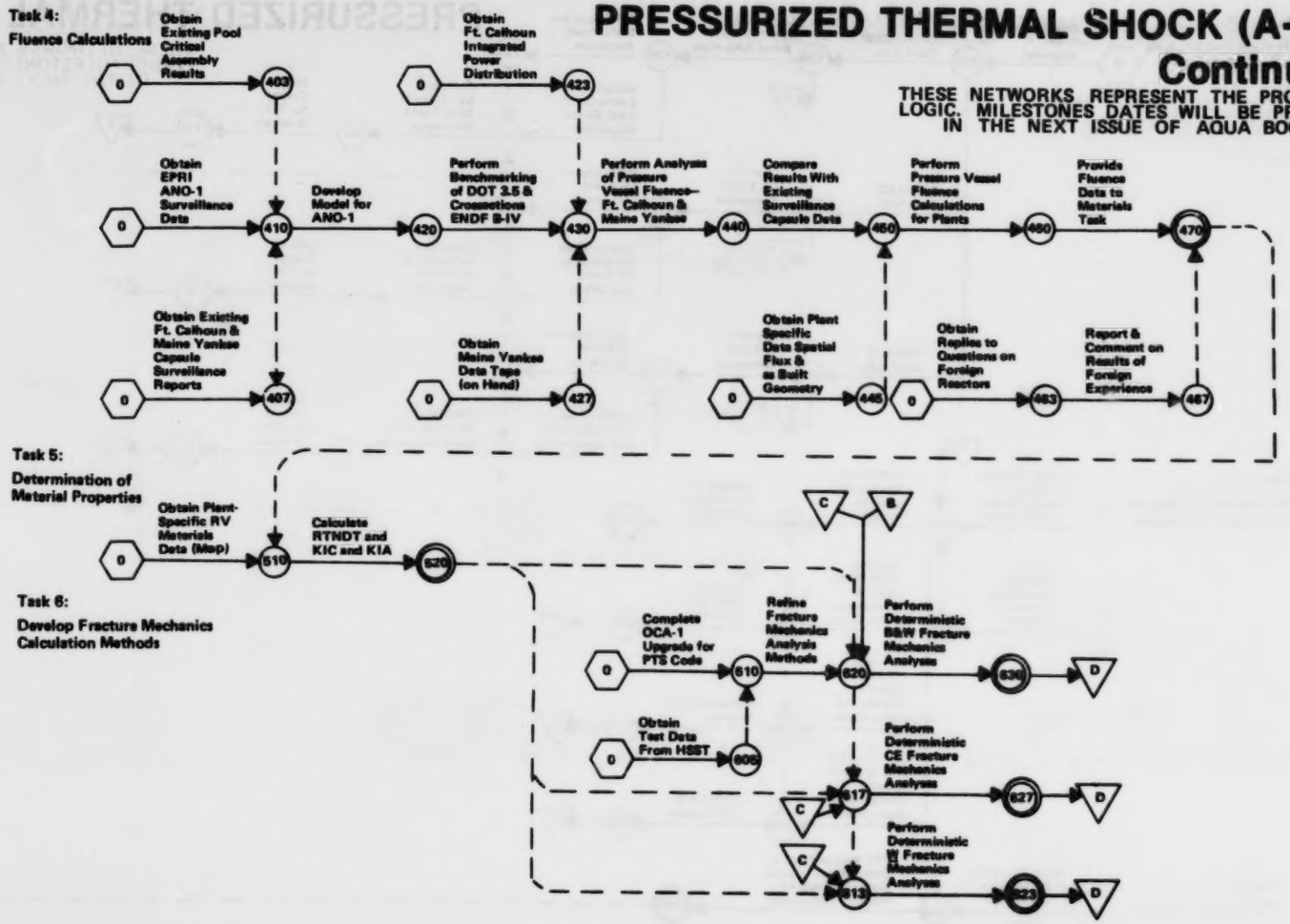
THESE NETWORKS REPRESENT THE PROGRAM LOGIC. MILESTONES DATES WILL BE PROVIDED IN THE NEXT ISSUE OF AQUA BOOK.



PRESSURIZED THERMAL SHOCK (A-49)

Continued

THESE NETWORKS REPRESENT THE PROGRAM LOGIC. MILESTONES DATES WILL BE PROVIDED IN THE NEXT ISSUE OF AQUA BOOK.



PRESSURIZED THERMAL SHOCK (A-49)
Continued

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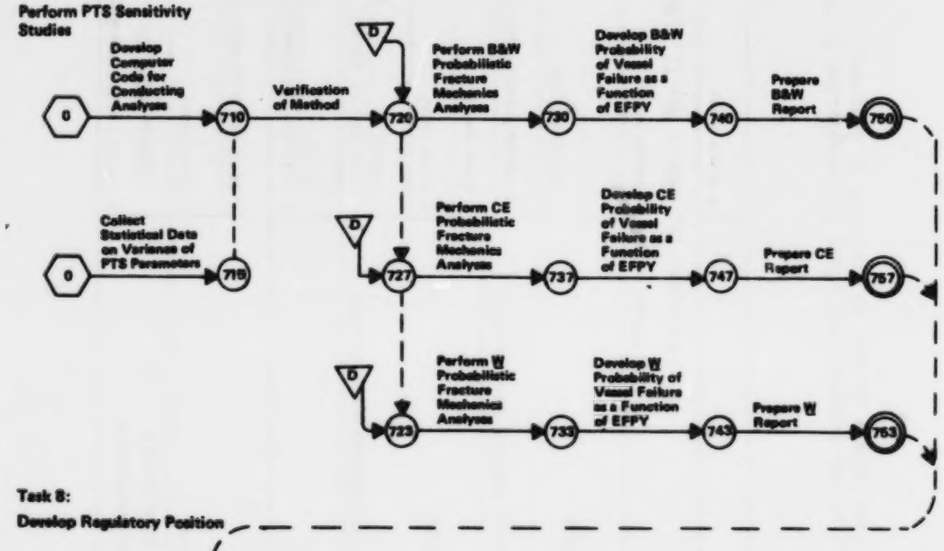
Task 7:
Perform PTS Sensitivity Studies
Develop Computer Code for Conducting Analyses
Verification of Method
Perform B&W Probabilistic Fracture Mechanics Analysis
Develop B&W Probability of Vessel Failure as a Function of EFPY
Prepare B&W Report
Collect Statistical Data on Variance of PTS Parameters
Perform CE Probabilistic Fracture Mechanics Analysis
Develop CE Probability of Vessel Failure as a Function of EFPY
Prepare CE Report
Perform W Probabilistic Fracture Mechanics Analysis
Develop W Probability of Vessel Failure as a Function of EFPY
Prepare W Report
Task 8:
Develop Regulatory Position
Assessment of Identified Corrective Measures
Formulate Regulatory Position
Complete DST Draft
Complete Division Review
Complete NRR Director Review
CRGR Review
CRGR Decision
Issue for Public Comment
Complete Incorporation of Comments
Complete Division Review
Complete NRR Director Review
CRGR Review
CRGR Decision
Issue Final NUREG

NRC FORM 335 (7-77) U.S. NUCLEAR REGULATORY COMMISSION

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16. ABSTRACT *(200 words or less)*

Provide an overview of the status of the progress and plans for resolution of the generic tasks addressing "Unresolved Safety Issues" as reported to Congress.

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AQUA BOOK

Manuscript Completed: August 1982
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Prepared for: OFFICE OF RESOURCE MANAGEMENT

Prepared by: OFFICE OF RESOURCE MANAGEMENT
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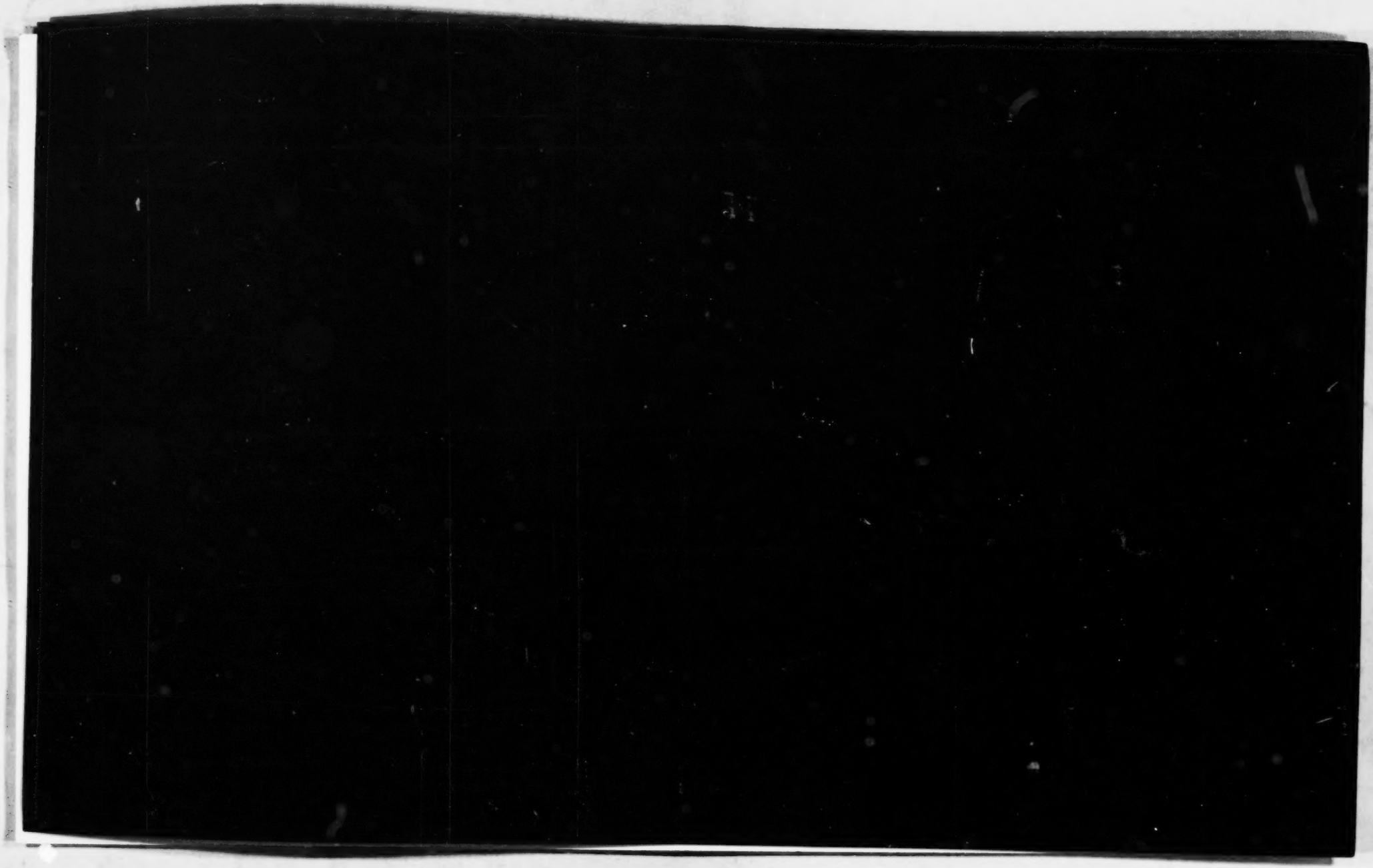

FOREWORD

THE "UNRESOLVED SAFETY ISSUES" SUMMARY IS DESIGNED TO PROVIDE THE MANAGEMENT OF THE NUCLEAR REGULATORY COMMISSION WITH A QUARTERLY OVERVIEW OF THE PROGRESS AND PLANS FOR COMPLETION OF GENERIC TASKS ADDRESSING "UNRESOLVED SAFETY ISSUES" REPORTED TO CONGRESS PURSUANT TO SECTION 210 OF THE ENERGY REORGANIZATION ACT OF 1974 AS AMENDED. THIS SUMMARY UTILIZES DATA COLLECTED FROM THE OFFICE OF NUCLEAR REACTOR REGULATION, OFFICE OF NUCLEAR REGULATORY RESEARCH, AND THE NATIONAL LABORATORIES AND IS PREPARED BY THE OFFICE OF RESOURCE MANAGEMENT.

THE DEFINITION OF WHAT CONSTITUTES COMPLETION OF AN UNRESOLVED SAFETY ISSUE (USI) HAS RECENTLY BEEN EXPANDED TO INCLUDE THE IMPLEMENTATION OF THE TECHNICAL RESOLUTION. THIS IS IN ACKNOWLEDGEMENT OF THE FACT THAT REAL SAFETY BENEFITS OCCUR ONLY AFTER THE IMPLEMENTATION HAS TAKEN PLACE. IMPORTANT ELEMENTS OF THIS IMPLEMENTATION PHASE ARE:

1. THE PROVISION OF A PUBLIC COMMENT PERIOD FOLLOWING THE ISSUANCE OF A DRAFT NUREG REPORT INCORPORATING THE STAFF'S TECHNICAL RESOLUTION FOLLOWED BY A DISCUSSION AND DISPOSITION OF THE COMMENTS RECEIVED IN A FINAL NUREG REPORT.

2. THE PROVISION FOR INCORPORATION OF THE TECHNICAL RESOLUTION INTO THE NRC'S REGULATIONS, STANDARD REVIEW PLAN, REGULATORY GUIDES, OR OTHER NRC OFFICIAL GUIDANCE OR REQUIREMENTS AS APPROPRIATE.

3. THE PROVISION FOR APPLICATION OF THE TECHNICAL RESOLUTION TO INDIVIDUAL OPERATING PLANTS IN THE FORM OF HARDWARE OR DESIGN CHANGES, TECHNICAL SPECIFICATION CHANGE, AND/OR CHANGE TO OPERATING PROCEDURES AS APPROPRIATE.

THE SCHEDULES IN THIS BOOK INCLUDE A MILESTONE AT THE END OF EACH ACTION PLAN WHICH REPRESENTS THE INITIATION OF THE IMPLEMENTATION PROCESS BOTH WITH RESPECT TO INCORPORATION OF THE TECHNICAL RESOLUTION IN THE NRC OFFICIAL GUIDANCE OR REQUIREMENTS AND ALSO THE APPLICATION OF CHANGES TO INDIVIDUAL OPERATING PLANTS. THE SCHEDULE FOR IMPLEMENTATION WILL NOT NORMALLY BE INCLUDED IN THE TASK ACTION PLAN(S) FOR THE RESOLUTION OF A USI SINCE THE NATURE AND EXTENT OF THE ACTIVITIES NECESSARY TO ACCOMPLISH THE IMPLEMENTATION CANNOT NORMALLY BE REASONABLY DETERMINED PRIOR TO THE DETERMINATION OF A TECHNICAL RESOLUTION. THE PROGRESS AND STATUS FOR IMPLEMENTATION OF UNRESOLVED SAFETY ISSUES FOR WHICH A TECHNICAL RESOLUTION HAS BEEN COMPLETED ARE REPORTED SPECIFICALLY IN A SEPARATE TABLE PROVIDED IN THIS SUMMARY.

KARL KNIEL, CHIEF OF THE GENERIC ISSUES BRANCH, DIVISION OF SAFETY TECHNOLOGY/NRR, IS RESPONSIBLE FOR MANAGING THE GENERIC TASKS INCLUDED IN THIS SUMMARY.

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ABBREVIATIONS

AAB:	Accident Analysis Branch (Former NRR Branch)
AB:	Administration Branch, Training and Administration Staff (IE)
AC:	Alternating Current
ACRS:	Advisory Committee on Reactor Safeguards
AD:	Assistant Director
ADB:	Analysis and Development Branch, Division of Reactor Safety Research (RES)
AEB:	Accident Evaluation Branch, Division of Systems Integration (NRR)
AEOD:	Office of the Analysis and Evaluation of Operational Data
AIF:	Atomic Industrial Forum
APTS:	Action Plan Tracking System
ARL:	Alden Research Laboratory
ASB:	Auxiliary Systems Branch, Division of Systems Integration (NRR)
ASME:	American Society of Mechanical Engineers
ASTM:	American Society of Testing Materials
ATWS:	Anticipated Transient Without Scram
B&W:	Babcock and Wilcox Company
BNC:	Brookhaven National Conservatory
BNL:	Brookhaven National Laboratory
BOP:	Balance of Plant
BWR:	Boiling Water Reactor
CE:	Combustion Engineering, Incorporated
CEB:	Chemical Engineering Branch, Division of Engineering (NRR)
CFR:	Code of Federal Regulations
CO:	Condensation Isolation
CP:	Construction Permit
CPB:	Core Performance Branch, Division of Systems Integration (NRR)
CR:	Contractor Report
CRGR:	Committee for Review of Generic Requirements
CSB:	Containment Systems Branch, Division of Systems Integration (NRR)
DC:	Direct Current
DE:	Division of Engineering (NRR)
DFO:	Division of Facility Operations (RES)
DHFS:	Division of Human Factors Safety (NRR)
DHRS:	Decay Heat Removal System
DL:	Division of Licensing (NRR)
DOE:	U. S. Department of Energy
DOR:	Division of Operating Reactors (Former NRR Division)
DRA:	Division of Risk Analysis (RES)
DSI:	Division of Systems Integration (NRR)
DSS:	Division of Systems Safety (Former NRR Division)
DST:	Division of Safety Technology (NRR)
E:	Engineering
EB:	Enforcement Branch, Enforcement and Investigations Staff (IE)
ECC:	Emergency Core Cooling
EEB:	Environmental Engineering Branch, Division of Engineering (NRR)
EFPY:	Effective Full-Power Years
EG&G:	Edgerton, Germeshausen & Grier
EP:	Emergency Preparedness
EPRI:	Electric Power Research Institute
EQB:	Equipment Qualification Branch, Division of Engineering (NRR)
FIN:	Financial
FIRL:	Franklin Institute Research Laboratories
FSI:	Fluid Structure Interaction
FSTF:	Full-Scale Test Facility
FW:	Feedwater
GB:	Geosciences Branch, Division of Engineering (NRR)
GE:	General Electric
GIB:	Generic Issues Branch, Division of Safety Technology (NRR)
HFEB:	Human Factors Engineering Branch, Division of Health, Siting and Waste Mgmt. (RES)
HFS:	Division of Human Factors Safety (NRR)
HSST:	Heavy Section Steel Technology
ICBR:	Instrumentation and Control Branch, Division of Facility Operations (RES)
ICSB:	Instrumentation and Control Systems Branch, Division of Systems Integration (NRR)
IE:	Office of Inspection and Enforcement
IEEE:	Institute of Electrical and Electronics Engineers
INEL:	Idaho Nuclear Engineering Laboratory
IREP:	Integrated Reliability Evaluation Program
ISI:	In-Service Inspection
LER:	Licensee Event Report
LGB:	Licensing Guidance Branch, Division of Safety Technology (NRR)
LLL:	Lawrence Livermore Laboratory
LOCA:	Loss-of-Coolant Accident
LPP:	Lead Plant Program
LTP:	Long Term Program
LWR:	Light-Water Reactor
MARK I-III:	Containment Types for Boiling Water Reactors
MEB:	Mechanical Engineering Branch, Division of Engineering (NRR)
MEBR:	Materials Engineering Branch, Division of Engineering Technology (RES)
MIT:	Massachusetts Institute of Technology
MMRB:	Metallurgy and Materials Branch, Division of Reactor Safety Research (RES)
MSLB:	Main Steam Line Break
MTEB:	Materials Engineering Branch, Division of Engineering (NRR)
NDE:	Non-Destructive Examination
NREP:	Neutron Resonance Escape Probability
NRL:	Naval Research Laboratory
NRR:	Office of Nuclear Reactor Regulation
NSS:	Nuclear Steam System
NUREG:	Nuclear Regulation

ABBREVIATIONS

OG: Owner's Group
OL: Operating License
ORAB: Operating Reactors Assessment Branch, Division of Licensing (NRR)
ORB: Operating Reactors Branch, Division of Licensing (NRR)
ORNL: Oak Ridge National Laboratory
OSD: Office of Standards Development (Former NRC Office)
OTSG: Once-Through Steam Generator
PASNY: Power Authority of the State of New York
PNL: Pacific Northwest Laboratory (Battelle)
PS: Plant Systems
PSB: Power Systems Branch, Division of Systems Integration (NRR)
PSU: Plant Systems Unit (AEOD)
PSYB: Plant Systems Branch (Former NRR Branch)
PTRB: Procedures and Test Review Branch, Division of Human Factors Safety (NRR)
PWR: Pressurized Water Reactor
Q: Quality Control or Assurances
RAB: Radiological Assessment Branch, Division of Systems Integration (NRR)
RCIC: Reactor Core Isolation Cooling
REF: Reference
RES: Office of Nuclear Regulatory Research
RFP: Request for Proposal
RHR: Residual Heat Removal
RPV: Reactor Pressure Vessel
PRA: Probability and Risk Assessment
RRAB: Reliability and Risk Assessments Branch (NRR)
RRRC: Regulatory Requirements Review Committee
RS: Reactor Safety (Former NRR Branch)
RSB: Reactor Systems Branch, Division of Systems Integration (NRR)
RSR: Reactor Safety Research
RSSMAP: Reactor Safety Study Methodology Application Program
SCC. Stress-Corrosion Cracking
SEB: Structural Engineering Branch, Division of Engineering (NRR)
SEP: Systematic Evaluation Program
SEPB: Systematic Evaluation Program Branch, Division of Licensing (NRR)
SER: Safety Evaluation Report
SG: Steam Generator
SGWH: Steam Generator Water Hammer
SNL: Sandia National Laboratory
SRP: Standard Review Plan
SRV: Safety Relief Valve
SSE: Safe Shutdown Earthquake
SSPB: Standards Special Projects Branch (NRR)
STP: Short-term Program
TAP: Task Action Plan
TH: Hot Leg Temperature
TM: Task Manager
TMI: Three Mile Island
USI: Unresolved Safety Issue
W: Westinghouse Electric Corporation
WH: Water Hammer
WUTA: Washington University Technology Associates

PROGRAM OVERVIEW

PROJECTED DATES FOR ISSUING NRC STAFF REPORTS

Key:
(1) LPP - Lead Plant Program
(2) LTP - Long Term Program
N/S - Not Scheduled

USI NO.	TITLE	MAY 21, 1982 AQUA BOOK	CURRENT	REMARKS
A-1,	Water Hammer	August 1, 1983	August 1, 1983	NUREG/CR-2781 was issued in July 1982. Development of a technical findings NUREG is underway. This NUREG will be used to develop recommendations to CRGR for resolution.
A-3, A-4, A-5,	Steam Generator Tube Integrity	March 30, 1983	March 30, 1983	The draft NUREG report has been reviewed by the technical branches and their comments incorporated. The report is being prepared for management review and then will be issued for public comment. The NRC has formed a Task Force to prepare proposed requirements.
A-7,	Mark I Long Term Program	Supplement- June 1, 1982	Supplement- August, 1982	This issue is considered complete since the final NUREG was issued July 1980 and confirmatory orders implementing the findings in NUREG 0661 were issued on January 13, 1981 for 22 operating BWRs with Mark I containments. However, BNL has reviewed three confirmatory issues for an SER supplement and issued a draft report for NRR review. The NRR supplement was issued August, 1982.
A-11,	Reactor Vessel Materials Toughness	October 15, 1982	October 15, 1982	NUREG-0774, Rev. 1, was completed and submitted for review. This revision addresses public comments received on the For Comment report. Issuance is expected in September 1982 thereby resolving this Unresolved Safety Issue.

PROGRAM OVERVIEW

PROJECTED DATES FOR ISSUING NRC STAFF REPORTS

USI NO.	TITLE	MAY 21, 1982 AQUA BOOK	CURRENT	REMARKS
A-12,	Steam Generator & Reactor Coolant Pump Supports	November 15, 1982	December 30, 1982	Meetings were held with EPRI and Industry groups in December 1980, to resolve use of fracture mechanics methods for qualification of supports. NRC reviewer's comments on the final NUREG-0577 were not received until mid-January 1982, causing a delay in issuing the final report. Comments made by EPRI and the AIF subcommittee on material requirements have been incorporated into the final NUREG.
A-17,	Systems Interactions	See Item II.C.3 in Action Plan Tracking System	See Item II.C.3 in Action Plan Tracking System	Work planned under TAP A-17 will be performed under TMI Action Plan item II.C.3, Systems Interaction. The status of this item is reported in the quarterly updates of the TMI Action Plan Tracking System (APTS).
A-39,	SRV Pool Dynamic Loads	Mark II-July 15, 1982 Mark III-July 15, 1982	Mark II-September 30, 1982 Mark III-October 15, 1982	NUREG-0763 on in-plant SRV tests was completed and issued in June 1981. NUREG-0783 on Pool Temperature Limits was issued on October 26, 1981. NUREG-0802 on SRV Quencher loads and the NUREG on suppression pool temperature limits have been prepared and discussed in an August 11, 1982 meeting with CRGR. Formal approval is pending.
A-40,	Seismic Design Criteria	May 15, 1983	July 15, 1983	A draft report presenting staff position has been prepared and is under management review.
A-43,	Containment Emergency Sump	July 15, 1983	July 15, 1983	NUREG/CR-2760 and NUREG/CR-2772 were issued in June 1982. NUREG/CR-2758 was issued in July 1982. A resolution NUREG is in the final stages of preparation and will undergo management concurrence.

PROGRAM OVERVIEW

PROJECTED DATES FOR ISSUING NRC STAFF REPORTS

USI NO.	TITLE	MAY 21, 1982 AQUA BOOK	CURRENT	REMARKS
A-44,	Station Blackout	October 15, 1983	October 15, 1983	Technical Assistance Contracts with ORNL and Sandia have been issued. A request for information on Diesel Generator reliability was sent to all licensees in July 1981; responses from licensees have been received. Draft contractor technical reports were provided in May 1982.
A-45,	Shutdown Decay Heat Removal Requirements	September 30, 1985	October 30, 1985	A Task Action Plan was issued on October 7, 1981, and revised on June 2, 1982. A technical assistance contract package was approved on April 9, 1982 and implemented with Sandia on May 10, 1982.
A-46,	Seismic Qualification of Equipment in Operating Plants	May 1, 1985	May 1, 1984	This program was redirected to be made compatibile with the Equipment Qualification Program Plan. The final Task Action Plan has been issued. Work has been initiated on all tasks with all contractors well into their subtasks.
A-47,	Safety Implications of Control Systems	N/S	N/S	A revised draft Task Action Plan was issued in April 1982 for staff concurrence and approval. Management approval of the Task Action Plan is expected by Sept. 1982. Technical Assistance contracts have been established with ORNL and INEL.
A-48,	Hydrogen Control Measures and Hydrogen Burns on Safety Equipment	N/S	N/S	A Task Manager has been selected. A draft Task Action Plan was issued on April 2, 1982 for NRC comments. Work has been suspended pending management decision on the scope of the TAP.
A-49,	Pressurized Thermal Shock	May 30, 1983	N/S	A-49 was designated by the Commission as an USI on December 28, 1981. The Task Action Plan was approved and issued on March 26, 1982. The draft short term position is under preparation and will be completed in September 1982.

USI'S FOR WHICH TECHNICAL RESOLUTION IS COMPLETE

USI NO.	TITLE	DATE COMPLETED	REPORTS PUBLISHED	IMPLEMENTATION STATUS	IMPLEMENTATION TASK MANAGER/ TELEPHONE
A-2,	ASYMMETRIC BLOWDOWN LOADS ON REACTOR PRIMARY COOLANT SYSTEMS	DEC. 1980	NUREG-0609	ALL PWR PLANT ASSESSMENTS FOR ASYMMETRIC LOADS HAVE BEEN RECEIVED AND ARE CURRENTLY BEING EVALUATED BY STAFF AND CONTRACTOR EG&G. AS A BASIS FOR THIS EVALUATION, STAFF IS IMPLEMENTING CRITERIA DEVELOPED AS PART OF THE TASK ACTION PLAN A-2. THESE CRITERIA ARE INCLUDED IN NUREG-0609. B&W AND CE OWNERS' GROUPS HAVE RESPONDED TO THE QUESTIONS RAISED AS A RESULT OF OUR INITIAL REVIEW OF THEIR SUBMITTALS ON THE ASSESSMENT OF THE PRIMARY SYSTEM COMPONENTS AND SUPPORTS TO ASYMMETRIC LOCA LOADS. THESE RESPONSES ARE CURRENTLY BEING REVIEWED AND THE REVIEW IS EXPECTED TO BE COMPLETED BY NOVEMBER 1983. THE WESTINGHOUSE OWNERS GROUPS EXPECT A RESOLUTION OF THIS ISSUE ON THE BASIS OF 'LEAK BEFORE BREAK' CRITERIA. AN INTERNAL REPORT, RECOMMENDING ACCEPTANCE OF THE 'LEAK BEFORE BREAK' CONCEPT ISSUED BY THE STAFF IN FEBRUARY 1981, HAS BEEN DISCUSSED WITH THE ACRS AND IS UNDER REVIEW BY NRR MANAGEMENT PRIOR TO REVIEW BY CRGR.	JIM SHEA 492-7231
A-6,	MARK I SHORT TERM PROGRAM	DEC. 1977	NUREG-0408	COMPLETE - ALL PLANT UNIQUE ANALYSES AND EQUIPMENT MODIFICATIONS AS REQUIRED WERE REVIEWED AND ACCEPTED AND APPROPRIATE TECHNICAL SPECIFICATION CHANGES WERE MADE.	- - -

USI'S FOR WHICH TECHNICAL RESOLUTION IS COMPLETE

USI NO.	TITLE	DATE COMPLETED	REPORTS PUBLISHED	IMPLEMENTATION STATUS	IMPLEMENTATION TASK MANAGER/ TELEPHONE
A-7,	MARK I LONG TERM PROGRAM	JULY 1980 AUGUST 1982	NUREG-0661 NUREG-0661, SUPPL. No. 1	LICENSEES ARE IN THE PROCESS OF DESIGNING AND INSTALLING MODIFICATIONS TO MEET THE COMMISSION'S ORDER DATE FOR EACH OPERATING PLANT. THE LICENSEES ARE IN THE PROCESS OF PREPARING PLANT-UNIQUE ANALYSIS FOR SUBMITTAL TO THE STAFF FOR POST IMPLEMENTATION AUDIT REVIEW FOR COMPLIANCE WITH THE ACCEPTANCE CRITERIA CONTAINED IN APPENDIX A TO NUREG-0661. THREE LICENSEES/APPLICANTS HAVE SUBMITTED THEIR PLANT UNIQUE ANALYSIS TO DATE.	BYRON SIEGEL 492-7534
A-8,	MARK II CONTAINMENT POOL DYNAMIC LOADS	AUGUST 1981	NUREG-0808	THE REQUIREMENTS RECOMMENDED IN NUREG-0808 ARE BEING IMPLEMENTED DURING THE OPERATING LICENSE REVIEW FOR EACH PLANT WITH A MARK II CONTAINMENT. THESE REQUIREMENTS WILL ALSO BE INCLUDED AS AN ADDITION TO THE APPROPRIATE SECTION OF THE STANDARD REVIEW PLAN.	
A-9,	ATWS	SEPT. 1980	NUREG 0460, VOL. 4 46FR57521	TWO VERSIONS OF THE PROPOSED ATWS RULE HAVE BEEN ISSUED FOR COMMENT; THE STAFF VERSION AND AN ALTERNATE VERSION OF THE RULE PREPARED AT THE DIRECTION OF COMMISSIONER HENDRIE. FOLLOWING THE COMMENT PERIOD AND CONSIDERATION OF PUBLIC COMMENTS, A FINAL RULE WILL BE PREPARED FOR COMMISSION CONSIDERATION. A TASK FORCE HAS BEEN ASSIGNED TO FORM A BASIS FOR THE FINAL RULE.	DAVID PYATT, RES 443-5960

USI'S FOR WHICH TECHNICAL RESOLUTION IS COMPLETE

USI NO.	TITLE	DATE COMPLETED	REPORTS PUBLISHED	IMPLEMENTATION STATUS	IMPLEMENTATION TASK MANAGER/ TELEPHONE
A-10,	BWR FEEDWATER NOZZLE CRACKING	NOV. 1980	NUREG-0619	RESPONSES FROM LICENSEES TO AN IMPLEMENTATION LETTER HAVE BEEN RECEIVED AND RECOMMENDED TREATMENT OF THESE RESPONSES HAVE BEEN SUBMITTED TO NRC MANAGEMENT. ADDITIONAL INFORMATION HAS BEEN REQUESTED OF LICENSEES. IMPLEMENTATION ACTION IS PROCEEDING. THIRTEEN PLANTS HAVE RECEIVED LETTERS ACCEPTING THEIR PROPOSED MODIFICATION PLANS. NINE PLANTS ARE BEING REVIEWED AND NEGOTIATIONS ARE UNDER WAY TO OBTAIN LICENSEE COMMITMENTS TO IMPLEMENT ALL NUREG-0619 GUIDANCE. LACROSSE, BIG ROCK POINT, AND DRESDEN 1 DO NOT HAVE SUSCEPTIBLE PLANT SYSTEM CONFIGURATIONS AND ARE CONSIDERED COMPLETE WITH REGARD TO THIS ACTION. HUMBOLDT BAY, BY VIRTUE OF ITS STATUS (SHUTDOWN, NO FORSEEABLE RESTART), IS ALSO CONSIDERED COMPLETE.	PAUL O'CONNOR 492-7215
A-24.	QUALIFICATION OF CLASS 1E SAFETY RELATED EQUIPMENT	AUG. 1981	NUREG-0588	THE NUREG REPORT WAS SUBMITTED TO THE PUBLISHER FOR PRINTING ON AUGUST 7, 1981. RESOLUTION OF PUBLIC COMMENTS AND ISSUANCE OF THE REVISED NUREG REPORT COMPLETES THE ACTIVITIES ON A-24. EQB HAS THE LEAD IN IMPLEMENTING THE POSITIONS IDENTIFIED IN THE REPORT.	JOSE' CALVO 492-8563
A-26,	REACTOR VESSEL PRESSURE TRANSIENT PROTECTION	SEPT. 1978	NUREG-0224	ALL PLANTS WERE REQUESTED TO PROVIDE AN OVERPRESSURE PREVENTION SYSTEM THAT WOULD BE USED WHENEVER THE PLANT WAS IN A COLD SHUTDOWN CONDITION. ALL PWRS IMPLEMENTED THEIR SYSTEMS WITH PRELIMINARY APPROVAL FROM THE NRC, AND A COMPLETE REVIEW TOOK PLACE ON A POST-IMPLEMENTATION BASIS. THE REVIEW CONSISTS OF THREE PARTS: A) ELECTRICAL ASPECTS: B) REACTOR SYSTEM ASPECTS; C) TECHNICAL SPECIFICATIONS. PLANTS REMAINING TO BE EVALUATED ARE BEING REVIEWED BY CONTRACT WITH EG&G. THIS INCLUDES NINE OF THE TWELVE REMAINING PLANTS. FOR THE OTHER THREE PLANTS, THE SEP'S HAVE BEEN COMPLETED AND ARE IN THE PROCESS OF BEING ISSUED.	AL DEAGAZIO 492-8945

USI'S FOR WHICH TECHNICAL RESOLUTION IS COMPLETE

USI NO.	TITLE	DATE COMPLETED	REPORTS PUBLISHED	IMPLEMENTATION STATUS	IMPLEMENTATION TASK MANAGER/ TELEPHONE
A-31,	RESIDUAL HEAT REMOVAL REQUIREMENTS	1978	NO FORMAL REPORT SRP 5.4.7 REVISION	RRRC APPROVED IMPLEMENTATION PLAN JANUARY 31, 1978. IMPLEMENTATION FOR OPERATING REACTORS NOT COMPLETE.	TOM NOVAK 492-7817
A-36,	CONTROL OF HEAVY LOADS NEAR SPENT FUEL	JULY 1980	NUREG-0612	IMPLEMENTATION REQUIREMENTS WERE ISSUED TO ALL LICENSEES BY LETTER DATED DECEMBER 22, 1980. THE LETTER REQUESTED; INTERIM ACTIONS TO BE COMPLETED IN 90 DAYS, A PHASE I ACTION (REPORT, CONFIRMATION AND JUSTIFICATION) IN SIX MONTHS AND PHASE II (SPECIFIC REQUIREMENTS) IN NINE MONTHS. MOST LICENSEES HAVE RESPONDED TO THE DECEMBER 22, 1980 GENERIC LETTER AND THEIR RESPONSES ARE BEING EVALUATED. ABOUT NINE PLANTS HAVE REQUESTED EXTENSIONS TO THE SEPTEMBER 22, 1981 RESPONSE DATE. SEVERAL PLANTS HAVE NOT YET ADEQUATELY RESPONDED TO THE EXTENT THAT AN EVALUATION MAY BE COMPLETED.	BUD REQUA 492-7877
A-42,	PIPE CRACKS IN BOILING WATER REACTORS	JULY 1980	NUREG-0313 REV. 1	IN FEBRUARY 1981, NUREG-0313, REV. 1 WAS ISSUED TO ALL HOLDERS OF BWR OPERATING LICENSES OR CONSTRUCTION PERMITS AND TO ALL APPLICANTS FOR OPERATING LICENSES. BY JULY 1, 1981, THE APPLICANTS/LICENSEES WERE TO PROVIDE THEIR PROGRAM FOR REPLACEMENT OF SERVICE SENSITITIVE LINES AND WELDS, THEIR PROGRAM FOR AUGMENTED INSERVICE INSPECTION, THEIR PROGRAM FOR IMPROVING THE WATER CHEMISTRY ENVIRONMENT AND INCOPORATION OF ADEQUATE LEAK DETECTION CAPABILITY. ALL LICENSEES BUT ONE HAVE RESPONDED AND THEIR REPLIES ARE BEING EVALUATED BY A TECHNICAL SUPPORT CONTRACTOR (INEL). INITIAL TER'S ARE SCHEDULED TO BE COMPLETED IN THE FIRST AND SECOND QUARTERS OF FY83.	DICK CLARK 492-7162

WATER HAMMER (A-1)

AS OF WEEK ENDING AUGUST 20, 1982

KEY PERSONNEL

TASK MANAGER
ALECK SERKIZ (DST) X24217

MPA ANALYST
B. ST. MARY X28330

TASK REVIEWERS

NAME	BRANCH
S. MCKAY	PTRB/DHFS
C. GRAVES	RSB/DSI

SCHEDULED COMPLETION

1978 ANNUAL REPORT	12-31-80
CURRENT	08-01-83

• PROBLEM DESCRIPTION

Since 1969 there have been over 150 incidents involving water hammers in BWRs and PWRs reported. The water hammers (or steam hammers) have involved steam generator feedrings and piping, the RHR system, ECC systems, and containment spray, service water, feedwater and steam lines. The incidents have been attributed to such causes as rapid condensation of steam pockets, steam-driven slugs of water, pump startup with partially empty lines, and rapid valve motion. Most of the damage reported has been relatively minor, involving pipe hangers and restraints; however, there have been several incidents which have resulted in piping and valve damage.

Task Action Plan, TAP A-1, subdivides resolution of this Unresolved Safety Issue (USI) into the following tasks:

Task 1: Water Hammer Summary Reports

Task 2: Resolution of Water Hammer Issue

Task 3: Water Hammer Position for Operating Reactors

Task 4: Water Hammer Safety Studies

Task 1 and 4 have been completed and the reports are on the network charts which follow. The near-term position for operating reactors is to permit continued operation since, as noted above, damage attributed to water hammer has been relatively minor. The long-term aspect of Task 3, is an integral part of the resolution of this issue.

Task 2 represents the resolution aspect of this USI and will result in the final resolution.

• RES INTERFACE INFORMATION

None.

• ACRS INTERFACE INFORMATION

A presentation was made to the ECCS Subcommittee on 12/03/81, plans for resolution of A-1 were presented.

• TECHNICAL ASSISTANCE CONTRACTS

Resolution efforts for A-1 are being conducted under FIN No. A6451 at INEL. Both EG&G-Idaho and Quadrex staff are being utilized.

	FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
	FY 81:			
	A6451	EG&G-Idaho	$120,000	$ 39,355
	FY 82:			
	A6451	EG&G-Idaho		
		10/81	$120,000	$ 44,069
		11/81	145,000	56,285
		12/81	305,000	67,613
		01/82	305,000	137,194
		02/82	305,000	178,590
		03/82	305,000	207,411
#		04/82	305,000	232,553
#		05/82	330,000	244,964
#		06/82	330,000	253,151

• POTENTIAL PROBLEMS

#Value/impact evaluations required for CRGR review package have not yet been started. Without changes in subcontracted efforts to complete A-1 ranking evaluation, 1-2 month schedule slippage is possible.

• STATUS SUMMARY

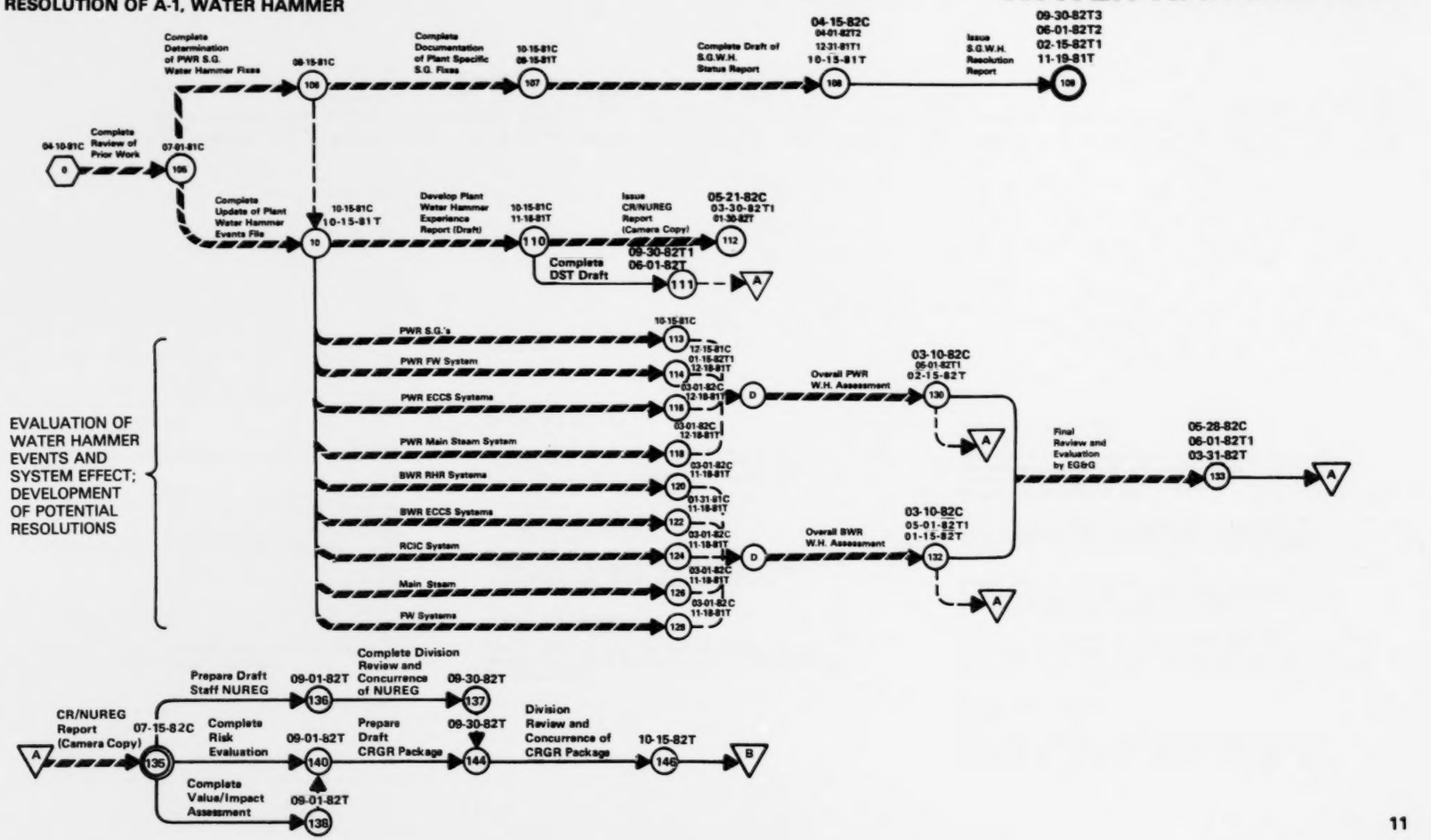
#NUREG/CR-2781, "Evaluation of Water Hammer Events in Light Water Reactor Plants," July 1982 was issued, thus completing Milestone No. 136, Task 2.0.

#Development of a technical findings NUREG is underway and was discussed by NRR staff on 08/03/82. This NUREG will be used to develop recommendations to the CRGR for resolution of USI A-1.

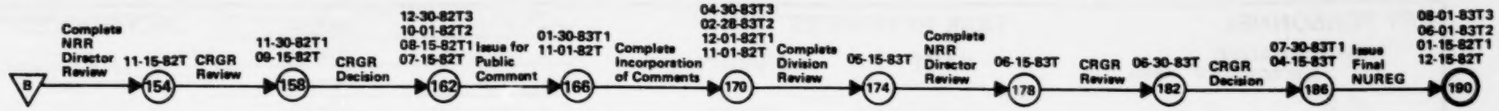
# INDICATES A CHANGE IN INFORMATION

TASK 2.0
RESOLUTION OF A-1, WATER HAMMER

WATER HAMMER (A-1)



WATER HAMMER (A-1) Continued



TASK 1 REPORTS

NUREG-0582, "Water Hammer in Nuclear Power Plants" #July 1979

Chapman, R.L., "Water Hammer Studies," EG&G Interim Rpt., CAAP-TR-053 (Rev. 1), July 1980 (Prepared for NRC Use)

#NUREG/CR-2059, "Compilation of Actual and Suspected Water Hammer Events in Nuclear Power Plants," May 1982.

TASK 2 REPORT

#NUREG/CR-2781, "Evaluation of Water Hammer Events in Light Water Reactor Plants," July 1982.

TASK 4.1 REPORT

Chapman, R.L., et al., "Review and Evaluation of Actual and Potential Water Hammer Events in Nuclear Plants," EG&G Interim Rpt., CAAP-TR-042 (Rev. 1), September 1979 (Prepared for NRC Use)

TASK 4.2 REPORT

Watkins, J.C., Berry, R.A., "A State-of-the-Art Literature Review of Water Hammer," EG&G Interm Interim Rpt., RE-A-70-044, April 1979 (Prepared for NRC Use)

TASK 4.3 REPORT

Saha, P., et al., "An Evaluation of Condensation - Induced Water Hammer in Preheat Steam Generators, "NUREG/CR-1606, September 1980

TASK 4.4 REPORTS

Williamson, R.L., "An Analysis Tool for Predicting the Transient Hydrodynamics Resulting from the Rapid Filling of Voided Piping Systems," EG&G Interim Rpt. RE-E-79-009, February 1979 (Prepared for NRC Internal Use)

Morton, D.K., "An Analytical Procedure for Performing Structural Analyses of Nuclear Piping Systems Subjected to Fluid Transients," EG&G Interim Rpt., RE-E-79-013, February (Prepared for NRC Internal Use)

Demmis, P.N., "An Investigation of the Steam Void Collapse Water Hammer Initiating Mechanism," EG&G Interim Rpt., RE-A-78-229, February 1979 (Prepared for NRC Internal Use)

Berry, R.A., "An Analysis Tool for Predicting Transient Hydrodynamics in Nuclear Piping Systems Containing Swing Check Valves," EG&G Rpt. RE-A-78-261 (Rev. 2), September 1979 (Prepared for NRC Internal Use)

TASK 4.5 REPORT

Saffell, B.F., "Supplemental Water Hammer Analysis and System Review," EG&G preliminary Rpt. EGG-CAAP-5133, July 1980 (Prepared for NRC for Use as a Preliminary or Working Document)

WEST., CE & B&W STEAM GENERATOR TUBE INTEGRITY (A-3, A-4, A-5)

AS OF WEEK ENDING AUGUST 20, 1982

KEY PERSONNEL

TASK MANAGER
J. STROSNIDER X36903

P. Norian for J. Strosnider

MPA ANALYST
B. ST. MARY X28330

TASK REVIEWERS

NAME	BRANCH
J. STROSNIDER	MEBR/RES
W. PASEDAG	AEB/DSI
C. PARSZEWSKI	CEB/DE
J. RAJAN	MEB/DE
B. TUROVLIN	CEB/DE
F. ODAR	ADB/RSR
F. AKSTULEWICZ	AEB/DSI
J. ZWOLINSKI	HFEB/DHFS
L. FRANK	MEBR

SCHEDULED COMPLETION

1978 ANNUAL REPORT — Early 1980

CURRENT — 03-30-83

• PROBLEM DESCRIPTION

Pressurized water reactor steam generator tube integrity can be degraded by corrosion induced wastage, cracking, reduction in tube diameter (denting) and vibration induced fatigue cracks. The primary concern is the capability of degraded tubes to maintain their integrity during normal operation and under accident conditions (LOCA or a main steam line break) with adequate safety margins.

Westinghouse and Combustion Engineering steam generator tubes have suffered degradation due to wastage and stress corrosion cracking. Both types of degradation have been decreased by conversion from phosphate to an all-volatile secondary water treatment. Degradation due to denting which leads to primary side stress corrosion cracking continues to be a problem.

B&W's once-through steam generators (OTSG's) were relatively free of trouble prior to the first tube leak incident at Oconee Unit 3 in July, 1976. Since then, all three Oconee units have experienced tube leak incidents. The leaks at the Oconee units are the result of cracks of unknown origins propagated in the circumferential direction by flow induced vibration and have been limited to tubes located adjacent to the open tube inspection lane.

A second form of degradation characterized as an erosion-cavitation phenomena has been observed at Oconee and other B&W units.

• RES INTERFACE INFORMATION

A. RES has funded, at the request of NRR, a major confirmatory program at PNL. The activity of this program consists of tests to verify the burst and cyclic strengths of degraded steam generator tubes and the leak rate data.
B. RES is funding a program addressing the factors which determine Inconel 600 susceptability to stress corrosion cracking in primary water. Metal condition, chemistry, temperature, stress and environment will be considered.

• ACRS INTERFACE INFORMATION

NONE

• TECHNICAL ASSISTANCE CONTRACTS

The following technical assistance contracts are generic in nature and will be applicable to the three Category "A" Technical Activities (A-3, A-4, and A-5) related to PWR steam generators.

A SANDIA — Provide statistical analysis of steam generator tube failures in operating reactors in order to establish the bases for the sampling plan for inservice inspection. Completed.
B BNL — Provide necessary computer code and perform parametric evaluation of effects of tube failures concurrent with MSLB. Completed.
C BNL — Provide technical consultation and assistance to review information in areas of water chemistory and corrosion analysis, stress and/or burst strength calculations. Completed.
D PNL — Provide cost/benefit evaluation of ISI plans. Completed.
E PNL — Evaluate environmental consequences of multiple tube failures concurrent with MSLB. Completed.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
2314	PNL	$75,000	$75,000
B2315	PNL	$95,000	$95,000

• POTENTIAL PROBLEMS

Management review and concurrence and CRGR approval are the only remaining actions necessary for issuance of the "For Comment" NUREG report.

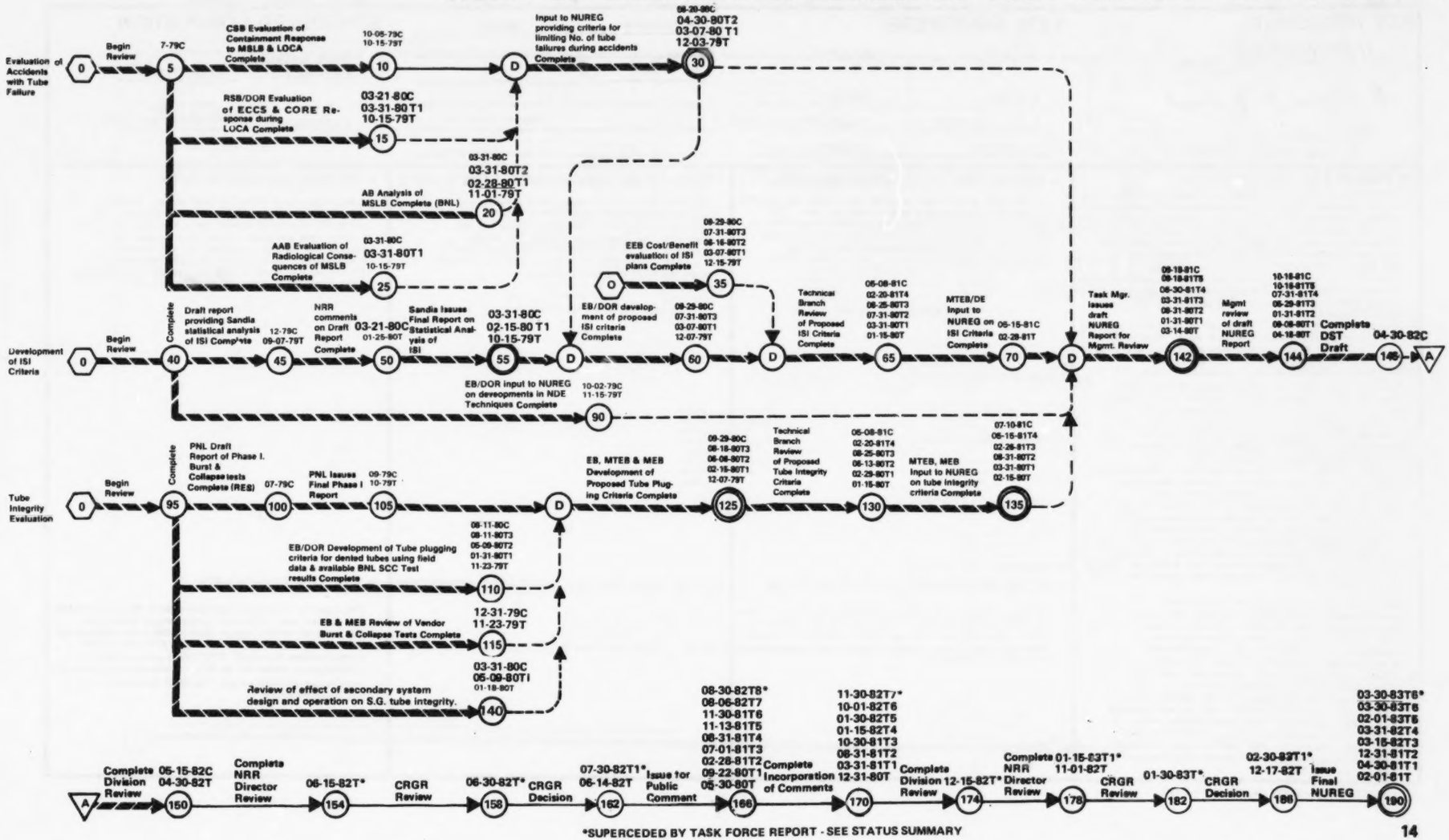
• STATUS SUMMARY

#The NRC has formed a Task Force to prepare its proposed requirements regarding steam generator tube integrity. These requirements will include new concerns resulting from the Ginna tube failure (such as loose parts in the secondary system and plant response to SG tube failures) and also corrosion related failure mechanisms. The recommendations prepared by the staff under USI A-3, 4, 5 were primarily concerned with corrosion mechanisms such as wastage and denting. Consequently, as discussed with the Commission on June 30, 1982, the requirements from the USI program will be incorporated in the overall set of requirements being developed to address tube failures.

#The proposed requirements were discussed with the Steam Generator Owners' Group on July 29, 1982. It is expected that a final report will be issued by the NRC in the Fall of 1982. USIs A-3, 4, 5 will be considered technically resolved when the overall raport has been approved and issued.

# INDICATES A CHANGE IN INFORMATION

WEST.. CE & B&W STEAM GENERATOR TUBE INTEGRITY (A-3, A-4,& A-5)



*SUPERCEDED BY TASK FORCE REPORT - SEE STATUS SUMMARY

MARK I CONTAINMENT LONG-TERM PROGRAM (A-7)

AS OF WEEK ENDING AUGUST 20, 1982

KEY PERSONNEL

TASK MANAGER
BYRON SIEGEL X27534

P. Norian for B. Siegel

MPA ANALYST
L. LANIER X28102

TASK REVIEWERS

NAME	BRANCH
B. SIEGEL	ORB/DL
T. SU, J. KUDRICK	CSB/DSI
R. BOSNAK	MEB/DE
F. SCHAUER	SEB/DE

SCHEDULED COMPLETION

1978 ANNUAL REPORT — OCTOBER 1979

CURRENT — JULY 1980C
#SUPPLEMENT 08-01-82C

• PROBLEM DESCRIPTION

During the conduct of a large scale testing program for an advanced design pressure-suppression containment system (Mark III) for BWRs, new suppression pool hydrodynamic loads associated with a postulated loss of coolant accident (LOCA) were identified which had not been explicitly included in the original design of the Mark I containment systems. In addition, experience at operating plants has indicated that the dynamic effects of safety-relief valve (SRV) discharges to the suppression pool could be substantial and should be reconsidered.

The results of the Mark I containment short-term program (STP) have provided assurance that for a "most probable load" the Mark I containment system of each operating BWR facility would maintain its integrity and functional capability during a postulated LOCA.

The need exists (1) to establish design basis LOCA loads which are appropriate for the life of the facility, and (2) to restore the originally-intended design safety margins for the containment systems or ensure that adequate design safety margin has been provided in the design of the containment system prior to issuance of an operating license.

The Mark I Owner's Group has initiated a comprehensive testing and evaluation program to define design basis loads for the Mark I containment system and to establish structural acceptance criteria which will assure margins of safety for the containment system which are equivalent to that which is currently specified in the ASME Boiler and Pressure Vessel Code. Also included in their program is an evaluation of the need for structural modifications and/or load mitigation devices to assure adequate Mark I containment system structural safety margins.

The NRC staff will evaluate the loads, load combinations, and associated structural acceptance criteria proposed by the Mark I Owner's Group prior to the conduct of plant-unique structural evaluations. The results of this evaluation will be documented in a generic Safety Evaluation Report. Publication of this report will constitute the resolution of this Technical Activity.

Implementation of the results of this generic review, although not a part of this task, will be accomplished by an NRC requirement that each affected utility perform a plant-unique structural evaluation of the containment system for their facility using the loads, loading combinations, and structural acceptance criteria approved by the NRC staff.

• RES INTERFACE INFORMATION

A 1/5 scale Mark I test program was conducted by LLL to provide confirmatory pool swell load data. The final test report was submitted to PSYB from RES on 03-10-78. Additional analyses of the test data were conducted by LLL to investigate 3-D pool swell effects and were submitted to PSYB in May 1979.

• ACRS INTERFACE INFORMATION

Meetings have been held with ACRS Subcommittee on Fluid Dynamics on 9/14/79 and 11/16/79 to discuss acceptance criteria. An additional meeting may be necessary to discuss changes to the criteria. A meeting with the full committee was held on 02-08-80.

• TECHNICAL ASSISTANCE CONTRACTS

BNL will assist in the review of the hydrodynamic load definition techniques (FIN A-3108). LLL has performed work on containment safety margins, fluid structure interaction effects, and column buckling. All of the LLL work was completed in FY 1979. The BNL work scope will be complete when the SER supplement is issued.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED

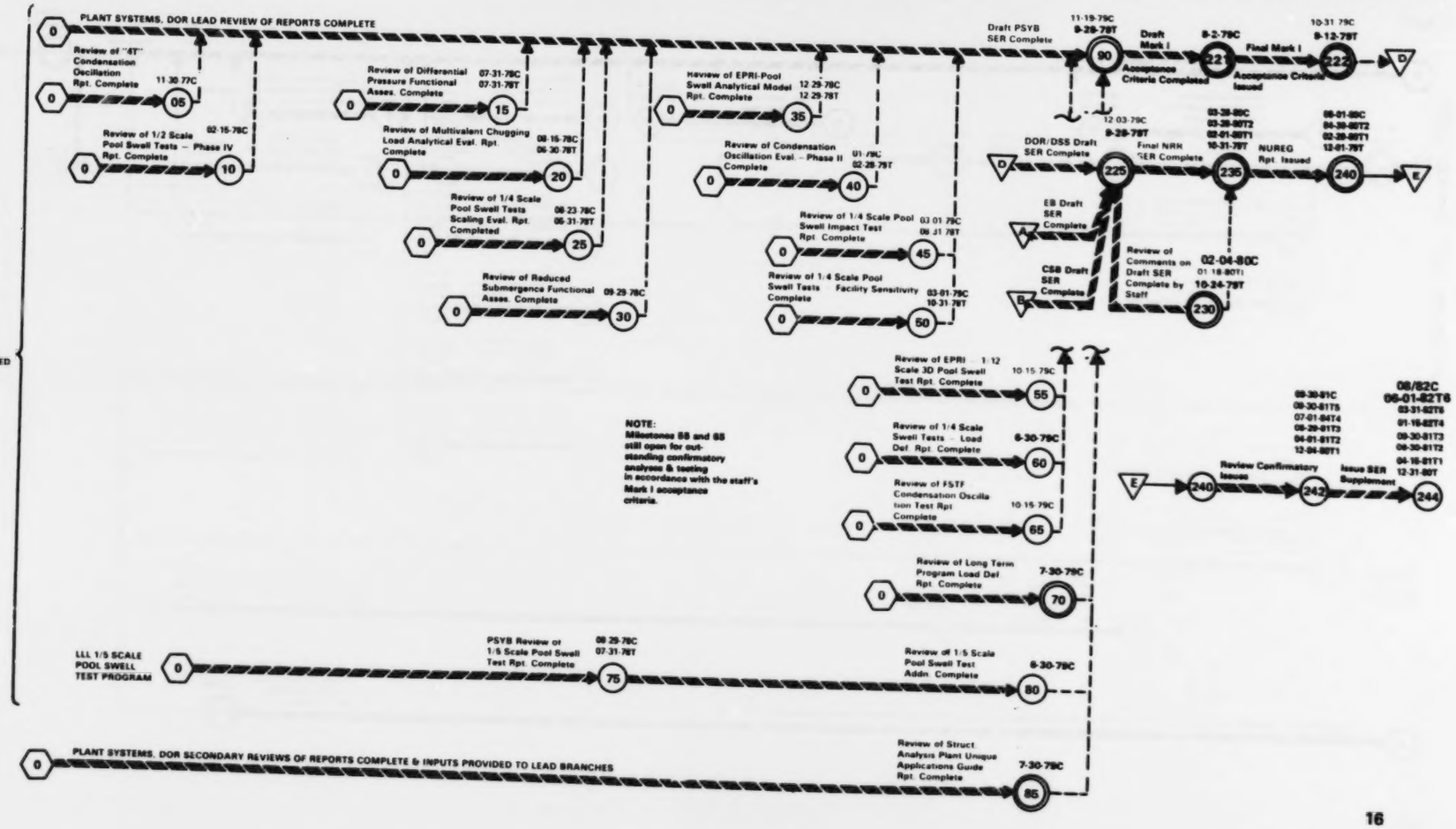
• POTENTIAL PROBLEMS

Comparisons of preliminary plant-unique analyses, using the proposed load definition techniques and the NRC acceptance criteria, with full-scale test data indicate that excessive stresses may be predicted because the loading conditions are too idealized. Confirmatory testing in the full-scale test facility (FSTF) may indicate a need for revised "condensation oscillation" loads, although none is expected.

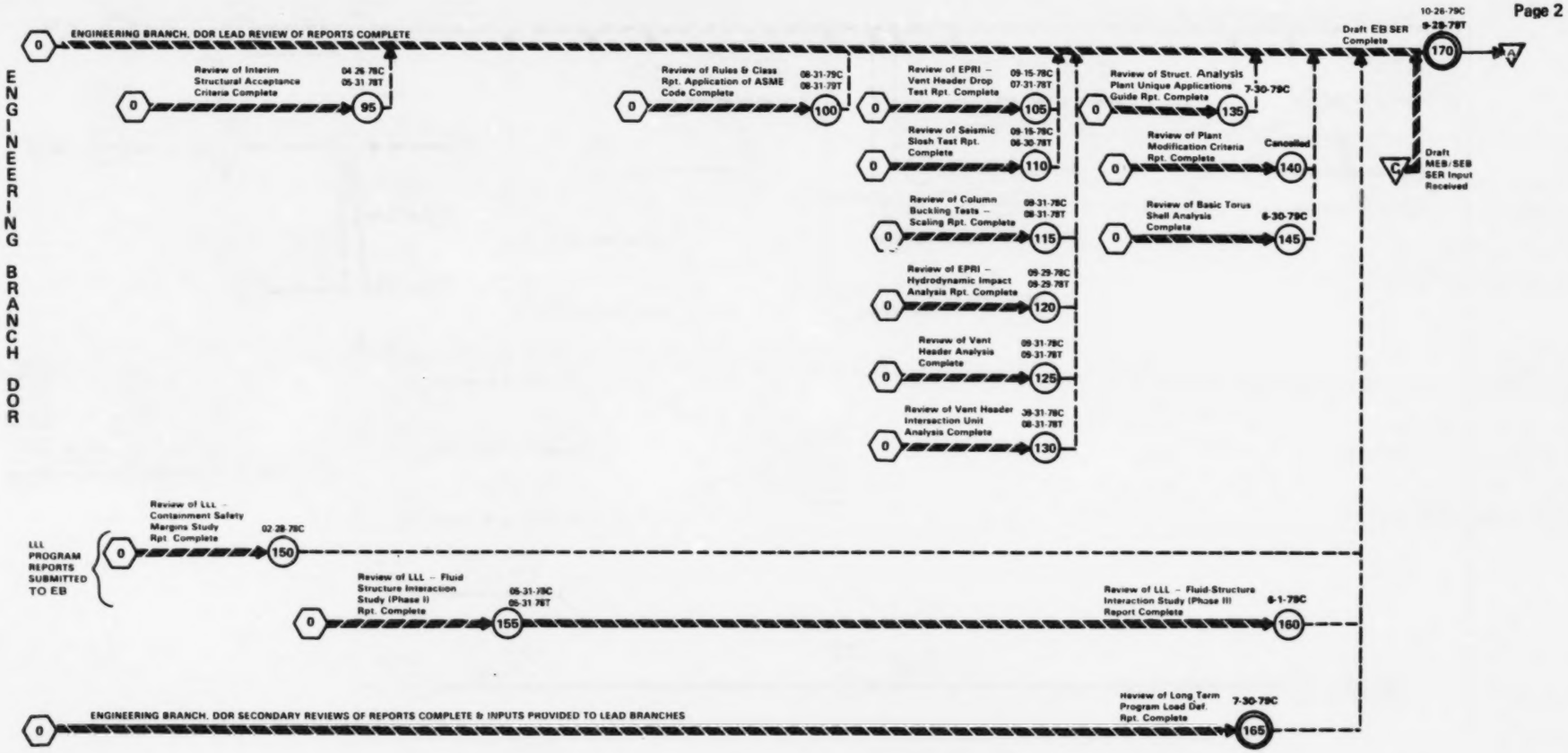
• STATUS SUMMARY

This issue is considered complete since the final NUREG was issued July, 1980, and confirmatory orders implementing the findings in NUREG-0661 were issued on January 13, 1981 for 22 operating BWRs with Mark I containments. However, BNL has reviewed three confirmatory issues for an SER supplement and issued a draft report for NRR review. The NRR supplement #was issued 08-01-82.

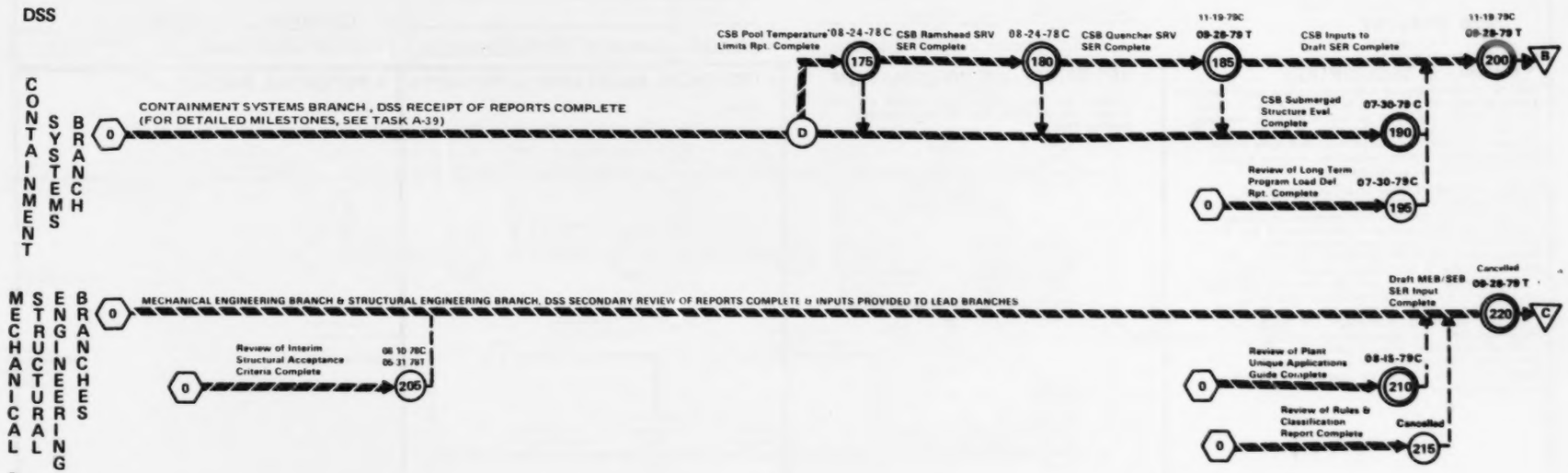
MARK I CONTAINMENT LONG-TERM PROGRAM (A-7)


PLANT SYSTEMS BRANCH DOR
BNL REPORTS SUBMITTED TO PSYB
PLANT SYSTEMS, DOR LEAD REVIEW OF REPORTS COMPLETE
Review of "6T" Condensation Oscillation Rpt. Complete 11-30-77C 05
Review of 1/2 Scale Pool Swell Tests — Phase IV Rpt. Complete 02-15-78C 10
Review of Differential Pressure Functional Asses. Complete 07-31-79C 07-31-78T 15
Review of Multivalent Chugging Load Analytical Eval. Rpt. Complete 08-15-78C 06-30-78T 20
Review of 1/4 Scale Pool Swell Tests Scaling Eval. Rpt. Completed 08-23-78C 06-31-78T 25
Review of Reduced Submergence Functional Asses. Complete 09-29-78C 30
Review of EPRI-Pool Swell Analytical Model Rpt. Complete 12-29-78C 12-29-78T 35
Review of Condensation Oscillation Eval. — Phase II Complete 01-79C 02-28-79T 40
Review of 1/4 Scale Pool Swell Impact Test Rpt. Complete 03-01-79C 06-31-78T 45
Review of 1/4 Scale Pool Swell Tests Complete Facility Sensitivity 03-01-79C 10-31-78T 50
Review of EPRI 1:12 Scale 3D Pool Swell Test Rpt. Complete 10-15-79C 55
Review of 1/4 Scale Swell Tests — Load Def. Rpt. Complete 6-30-79C 60
Review of FSTF Condensation Oscillation Test Rpt. Complete 10-15-79C 65
Review of Long Term Program Load Def. Rpt. Complete 7-30-79C 70
LLL 1/5 SCALE POOL SWELL TEST PROGRAM
PSYB Review of 1/5 Scale Pool Swell Test Rpt. Complete 08-29-78C 07-31-78T 75
Review of 1/5 Scale Pool Swell Test Addn. Complete 6-30-79C 80
PLANT SYSTEMS, DOR SECONDARY REVIEWS OF REPORTS COMPLETE & INPUTS PROVIDED TO LEAD BRANCHES
Review of Struct Analysis Plant Unique Applications Guide Rpt. Complete 7-30-79C 85
NOTE: Milestones 55 and 65 still open for outstanding confirmatory analyses & testing in accordance with the staff's Mark I acceptance criteria.
Draft PSYB SER Complete 11-19-79C 9-28-79T 90
Draft Mark I Acceptance Criteria Completed 8-2-79C 221
Final Mark I Acceptance Criteria Issued 10-31-79C 9-12-79T 222
D
DOR/DSS Draft SER Complete 12-03-79C 9-28-79T 225
Final NRR SER Complete 03-28-80C 03-28-80T2 02-01-80T1 10-31-79T 235
NUREG Rpt. Issued 08-01-80C 04-28-80T2 02-28-80T1 12-01-79T 240
E
EB Draft SER Complete
CSB Draft SER Complete
Review of Comments on Draft SER Complete by Staff 02-04-80C 01-18-80T1 10-24-79T 230
08/82C 06-01-82T6 03-31-82T5 01-15-82T4 09-30-81T3 08-30-81T2 04-15-81T1 12-31-80T
Review Confirmatory Issues 240
Issue SER Supplement 242 244

MARK I CONTAINMENT LONG-TERM PROGRAM (A-7)



MARK I CONTAINMENT LONG-TERM PROGRAM (A-7)


DSS
CONTAINMENT SYSTEMS BRANCH
CONTAINMENT SYSTEMS BRANCH, DSS RECEIPT OF REPORTS COMPLETE (FOR DETAILED MILESTONES, SEE TASK A-39)
CSB Pool Temperature Limits Rpt. Complete 08-24-78C
175
CSB Ramshead SRV SER Complete 08-24-78C
180
CSB Quencher SRV SER Complete 11-19-79C
185
CSB Inputs to Draft SER Complete 11-19-79C
200
CSB Submerged Structure Eval. Complete 07-30-79C
190
Review of Long Term Program Load Def. Rpt. Complete 07-30-79C
195
MECHANICAL & STRUCTURAL ENGINEERING BRANCHES
MECHANICAL ENGINEERING BRANCH & STRUCTURAL ENGINEERING BRANCH, DSS SECONDARY REVIEW OF REPORTS COMPLETE & INPUTS PROVIDED TO LEAD BRANCHES
Review of Interim Structural Acceptance Criteria Complete 08-10-78C 06-31-79T
205
Draft MEB/SEB SER Input Complete Cancelled 09-28-79T
220
Review of Plant Unique Applications Guide Complete 08-15-79C
210
Review of Rules & Classification Report Complete Cancelled
215

REACTOR VESSEL MATERIALS TOUGHNESS (A-11)

AS OF WEEK ENDING AUGUST 20, 1982

KEY PERSONNEL

TASK MANAGER
RICHARD E. JOHNSON X24715

Richard E. Johnson

MPA ANALYST
#L. LANIER X28012

TASK REVIEWERS

NAME	BRANCH
W.S. HAZELTON	MTEB/DE
R.W. KLECKER	MTEB/DE
J. STROSNIDER	MEBR/RES
P.N. RANDALL	MEBR/RES
C.S. HINSON	RAB/DSI
C. SERPAN	MEBR/RES

SCHEDULED COMPLETION

1978 ANNUAL REPORT	July 1979
CURRENT	10-15-82

• PROBLEM DESCRIPTION

Because the possibility of failure of nuclear reactor pressure vessels designed to the ASME Boiler and Pressure Vessel Code is remote, the design of nuclear facilities does not provide protection against reactor vessel failure. However, as plants accumulate more and more service time, neutron irradiation reduces the material fracture toughness and initial safety margins.

Results from reactor vessel surveillance programs indicate that up to approximately 20 operating PWRs will have beltline materials with marginal toughness, relative to the requirements of Appendices G and H of 10 CFR Part 20, after comparatively short (approximately 10 EFPY) periods of operation. For most plants now in the licensing process, current criteria, together with the materials currently employed, are adequate to ensure suitable safety margins for reactor vessels throughout their design lives. However, a few plants under licensing review have reactor vessels that have been identified as having the potential for marginal fracture toughness within their design lives; these vessels will have to be reevaluated in the light of the new criteria for long term acceptability.

• RES INTERFACE INFORMATION

Fracture test data was provided by RES and incorporated into NUREG-0744; RES has reviewed NUREG-0744.

• ACRS INTERFACE INFORMATION

#Copies of NUREG-0744, Rev. 1, will be forwarded to the ACRS when issued.

• TECHNICAL ASSISTANCE CONTRACTS

NONE

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
B6486	W.U.T.A.	$103,000	$ 83,088
B0737	ORNL	$415,000	$331,375
A6419	INEL	$100,000	$ 89,740
B6809	NRL	$ 95,650	$ 95,650

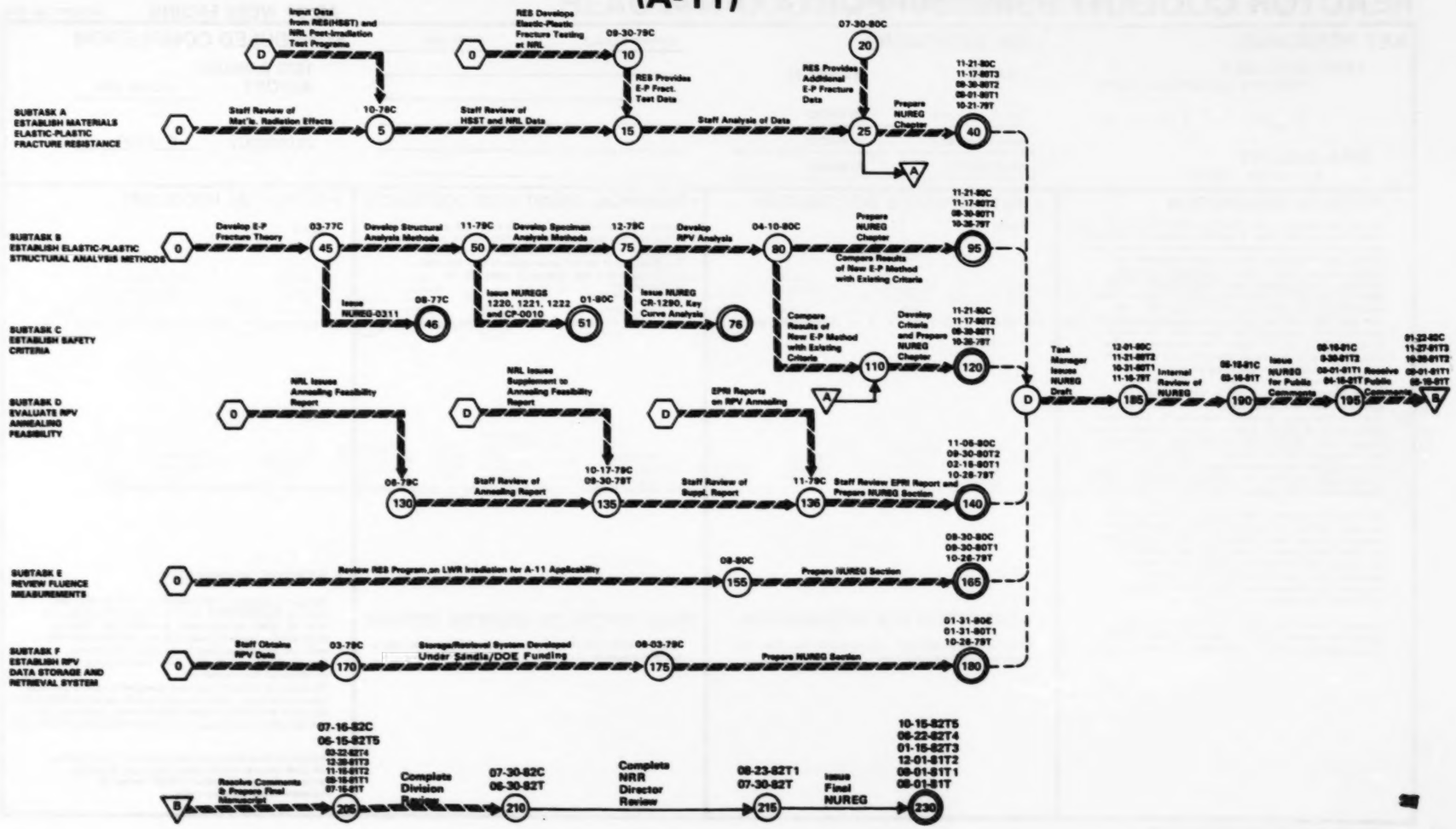
• POTENTIAL PROBLEMS

NONE

• STATUS SUMMARY

#NUREG-0744, Rev. 1, was completed and submitted for review. It incorporates formal public comments received, NRC staff resolution of comments, and changes to the NUREG resulting from the comments. After Rev. 1 has been reviewed by the Division, it will be submitted to the CRGR for information and then printed and issued. Those actions will complete Task A-11.

REACTOR VESSEL MATERIALS TOUGHNESS (A-11)



FRACTURE TOUGHNESS OF STEAM GENERATOR AND REACTOR COOLANT PUMP SUPPORTS (A-12)

AS OF WEEK ENDING AUGUST 20, 1982

KEY PERSONNEL

TASK MANAGER
RICHARD E. JOHNSON (GIB) X24715

Richard E. Johnson

MPA ANALYST
L. LANIER X28102

TASK REVIEWERS

NAME	BRANCH
DAVE SELLERS	MTEB(DE)
NEIL RANDALL	MTEB(DE)
DAVE SMITH	MTEB(DE)
BARRY ELLIOT	MTEB(DE)

SCHEDULED COMPLETION

1978 ANNUAL REPORT — AUGUST 1979

CURRENT — # 12-30-82

• PROBLEM DESCRIPTION

During the course of the licensing action for North Anna Power Station Unit Nos. 1 and 2 a number of questions were raised as to the potential for lamellar tearing and low fracture toughness of the steam generator and reactor coolant pump support materials for those facilities. Two different steel specifications (ASTM A36-70a and ASTM A572-70a) covered most of the material used for these supports. Toughness tests, not originally specified and not in the relevant ASTM specifications, were made on those heats for which excess material was available. The toughness of the A36 steel was found to be adequate, but the toughness of the A572 steel was relatively poor at an operating temperature of 80°F. In the case of North Anna Unit Nos. 1 and 2, the applicant has agreed to raise the temperature of the ASTM A572 beams in the steam generator supports to a minimum temperature of 225°F prior to reactor coolant system pressurization to levels above 1000 psig. Auxiliary electrical heat will be supplied as necessary to supplement the heat derived from the reactor coolant loop to obtain the required operating temperature of the support materials.

Since similar materials and designs have been used on other nuclear plants, the concerns regarding the supports for the North Anna facilities are applicable to other PWR plants. It was therefore necessary to reassess the fracture toughness of the steam generator and reactor coolant pump support materials for all operating PWR plants and those in CP and OL review.

Lamellar tearing may also be a problem in those support structures similar in design to North Anna. This possibility will be investigated on a generic basis.

• RES INTERFACE INFORMATION

NONE

• ACRS INTERFACE INFORMATION

Copies of NUREG-0577 will be forwarded to the ACRS when issued in revised and final form.

• TECHNICAL ASSISTANCE CONTRACTS

Sandia Laboratories has provided expertise in evaluating the potential for lamellar tearing and low fracture toughness of the support materials. The final input to the generic resolution has been submitted and is included as an appendix to NUREG-0577.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED

Contract assistance for preparation of final report from FIRL is being managed by DL.

• POTENTIAL PROBLEMS

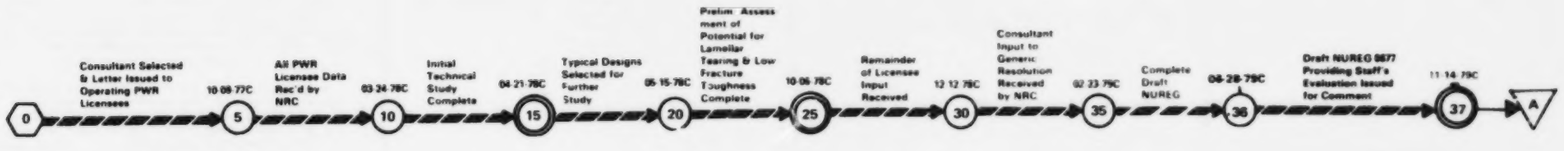
NONE

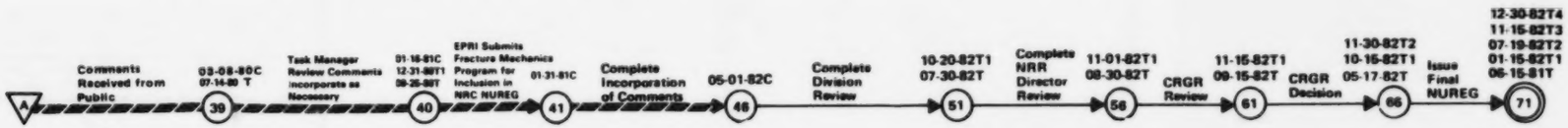
• STATUS SUMMARY

Review procedures and evaluation criteria were set forth in NUREG-0577 and the augmenting letters of May 19, 1980 (to licensees), and May 20, 1980 (to applicants). Industry response was adverse, and the staff has been requested to consider alternatives, first by EPRI and by the AIF Subcommittee on Material Requirements.
The scope of USI A-12 has been changed to reflect staff and industry concerns. Major changes include removing bolting from the scope and limiting applicability to steam generator and primary coolant pump supports as originally specified in the TAP.

Comments have been incorporated into final NUREG and final manuscript has been prepared. Schedule has been changed to reflect CRGR review of implementation requirements.

FRACTURE TOUGHNESS OF STEAM GENERATOR AND REACTOR COOLANT PUMP SUPPORTS (A-12)





SYSTEMS INTERACTIONS IN NUCLEAR POWER PLANTS (A-17)

AS OF WEEK ENDING AUGUST 20, 1982

KEY PERSONNEL

TASK MANAGER
PAUL NORIAN (Acting) X27487

Paul Norian

MPA ANALYST
L. LANIER X28102

TASK REVIEWERS

NAME BRANCH

NO ASSIGNED REVIEWERS

SCHEDULED COMPLETION

1978 ANNUAL REPORT Phase I - 09-79

CURRENT N/A

• PROBLEM DESCRIPTION

The design of a nuclear power plant is accomplished by groups of engineers and scientists organized into engineering disciplines and into scientific disciplines. The reviews performed by the designers include interdisciplinary reviews to assure the functional compatibility of the plant structures, systems, and components. Safety reviews and accident analyses provide further assurance that system functional requirements will be met. These reviews include failure mode analyses.

The NRC review and evaluation of safety systems is accomplished in accordance with the Standard Review Plan (SRP) which assigns primary and secondary review responsibilities to organizational units arranged by plant systems or by disciplines. Each element of the SRP is assigned to an organizational unit for primary responsibility and, where appropriate, to other units for secondary responsibilities.

Thus, the design and analyses by the plant designers, and the subsequent review and evaluation by the NRC staff take into consideration the interdisciplinary areas of concern and account for systems interaction to a large extent. Furthermore, many of our regulatory criteria are aimed at controlling the risks from systems interactions. Examples include the single failure criterion and separation criteria.

Nevertheless, there is some question regarding the interaction of various plant systems, both as to the supporting roles such systems play and as to the effect one system can have on other systems, particularly with regard to whether actions or consequences could adversely affect the presumed redundancy and independence of safety systems.

The problem to be resolved by this task is to identify where the present design, analysis, and review procedures may not acceptably account for potentially adverse systems interaction and to recommend the regulatory action that should be taken to rectify deficiencies in the procedures.

• RES INTERFACE INFORMATION

The Division of Risk Analysis has been consulted during the development and execution of this plan. The Division of Engineering Technology will participate in the development of a Regulatory Guide for Systems Interaction Analyses.

• ACRS INTERFACE INFORMATION

The review is under the cognizance of the ACRS Subcommittee on Plant Arrangements. A meeting was held 02-20-80 in Washington, D.C. with the Subcommittee which recommended a demonstration of Phase I methodology before proceeding much further.

A meeting was held 02-26-82 with the ACRS Subcommittee on Indian Point 3 to describe the PASNY systems interaction program on Indian Point 3. On 03-08-82, the ACRS documented their judgement that the PASNY plan was responsive.

• TECHNICAL ASSISTANCE CONTRACTS

Phase I work was performed by Sandia National Laboratories under FIN # A-1113(9) by NRR for $219K and under FIN # A-1084(8) by OSD for $320K. Expenditures were $197K for FIN # A-1113(9) and $320K for FIN # A-1084(8).

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
# A0403	LLNL	$200,000	$46,000
#A3365	BNL	$ 15,000	$15,000
# B2329	PNL	$ 10,000	$10,000
A1295	SANDIA	$ 27,000	$16,000
A-405	LLNL	$153,000	$129,000
A-3365	BNC	$ 80,000	$ 80,000
B-2330	PNL	$ 80,000	$ 80,000

• POTENTIAL PROBLEMS

LWRs for implementation of interim guidance have not yet been selected.

• STATUS SUMMARY

The anticipated Phase II program will not be pursued as an USI. Work originally planned under TAP A-17 will now be performed under TMI Action Plan Item II-C-3, Systems Interaction. This activity will now be managed by the Systems Interaction Section, Reliability and Risk Assessment Branch, DST. The Section Leader is F. Coffman, X27487 The status of this item is reported in the quarterly updates of the TMI Action Plan Tracking System (APTS).

Commission briefing scheduled for 06/82. Interim guidance for implementation on selected plants is scheduled for completion 08/82.

#The Chairman was briefed on 06/25/82 concerning the status of the systems interaction program and how it might proceed.

#Stop work orders were issued on contracts A-3365 and B-2329.

#The documentation of a matrix-digraph method is expected to be issued 11/82, under contract A-4030.

#The initial guidance document is expected to be issued 11/82.

# INDICATES A CHANGE IN INFORMATION

DETERMINATION OF SAFETY RELIEF VALVE (SRV) POOL DYNAMIC LOADS AND TEMPERATURE LIMITS FOR BWR CONTAINMENT (A-39)

AS OF WEEK ENDING AUGUST 20, 1982

KEY PERSONNEL

TASK MANAGER
T.M. SU (GIB) X27477

MPA ANALYST
M. ANDERSON X27021

TASK REVIEWERS

NAME	BRANCH

SCHEDULED COMPLETION

1978 ANNUAL REPORT — October 1979

CURRENT
- MARK I - 02-29-80C
- # MARK II - 09-30-82
- # MARK III - 10-15-82

• PROBLEM DESCRIPTION

Operation of BWR primary system pressure relief valves can result in hydrodynamic loads on the suppression pool retaining structures or structures located within the pool. These loads result from:

1. Initial vent clearing of a relief valve pipe; and
2. Steam quenching due to locally high pool temperature.

• RES INTERFACE INFORMATION

NONE

• ACRS INTERFACE INFORMATION

Discussions with the ACRS Subcommittee on Mark III SRV loads and Kuoshong inplant test results was presented on September 25, 1981.

• TECHNICAL ASSISTANCE CONTRACTS

BNL has completed Mark II T-Quencher and Mark III X-Quencher review and is preparing input to the NUREG report.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
A3088	BNL	FY 79 $65,000 FY 80 $60,000	
A3321	BNL	FY 80 $35,000 FY 81 $70,000	

• POTENTIAL PROBLEMS

NONE

• STATUS SUMMARY

Task A-39 Mark I effort completed in February 1980.

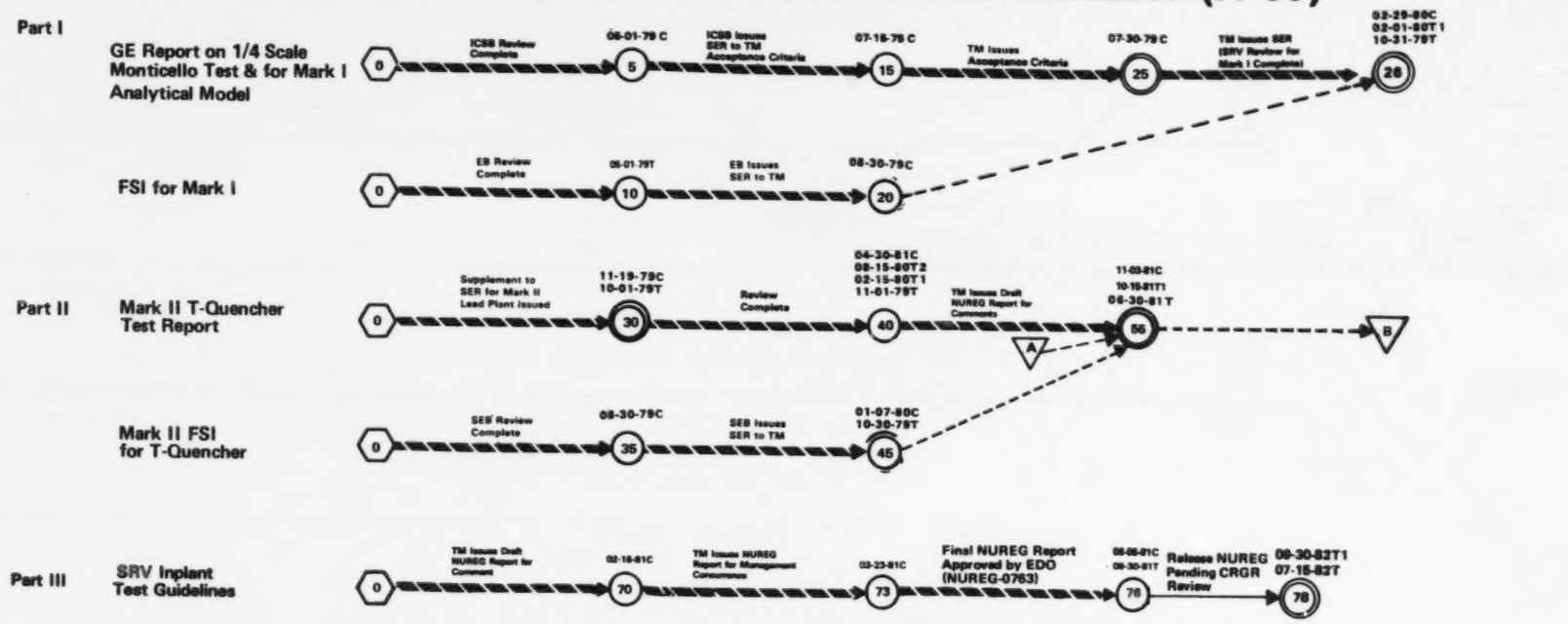
NUREG report for inplant SRV test was approved by EDO in June 1981. Issuance of the report, however, is pending CRGR approval.

NUREG report for suppression pool temperature limit was approved by EDO in 10/81. Issuance of the report is pending CRGR approval.

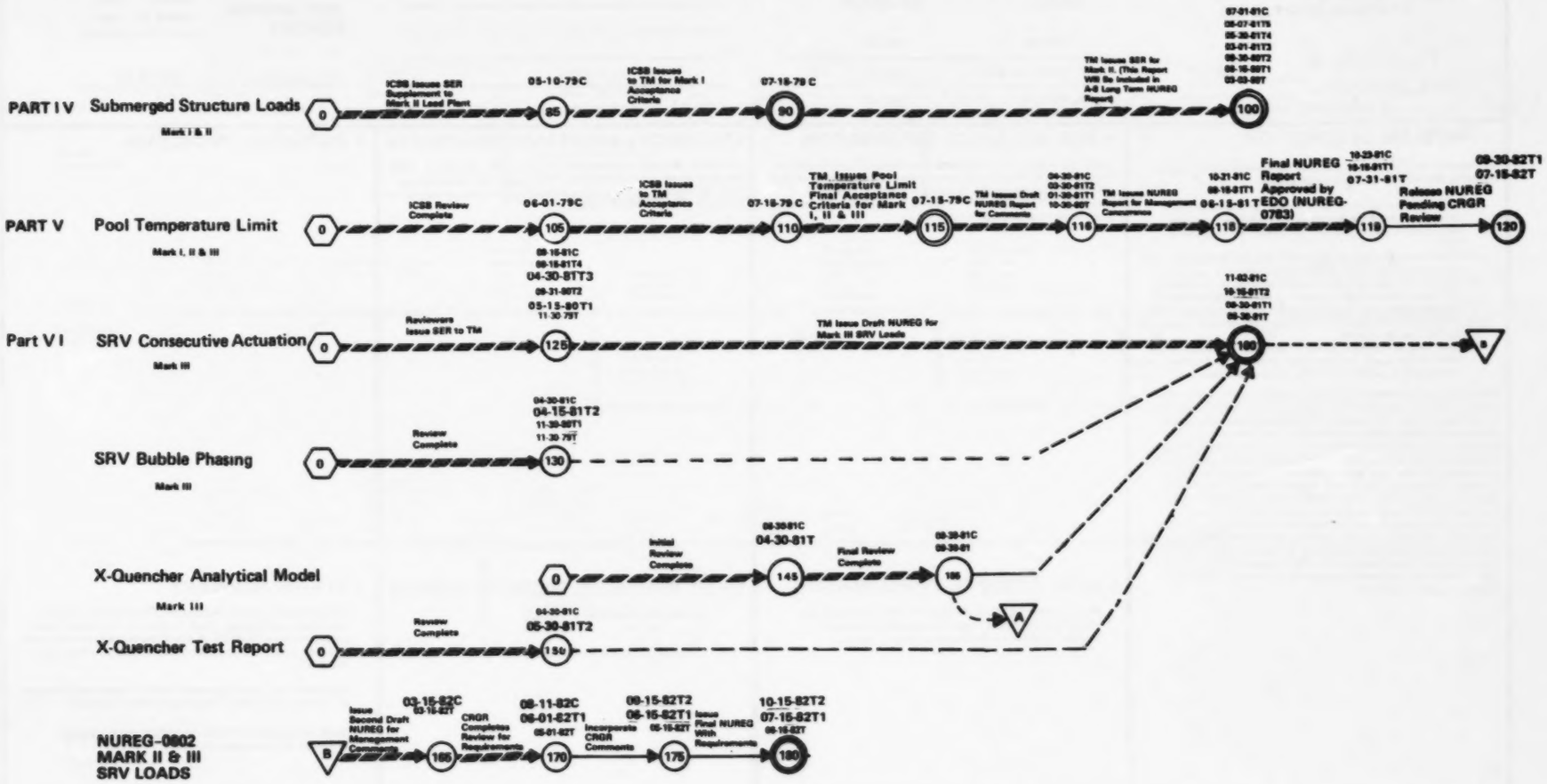
Draft NUREG report for Mark II and III SRV loads is complete and NRR management has concurred.

A meeting was held with the CRGR on August 11, 1982. CRGR formal approval is pending.

DETERMINATION OF SAFETY RELIEF VALVE (SRV) POOL DYNAMIC LOADS AND TEMPERATURE LIMITS FOR BWR CONTAINMENT (A-39)



DETERMINATION OF SAFETY RELIEF VALVE (SRV) POOL DYNAMICS LOADS AND TEMPERATURE LIMITS FOR BWR CONTAINMENT (A-39) Continued



SEISMIC DESIGN CRITERIA - SHORT TERM PROGRAM (A-40)

AS OF WEEK ENDING AUGUST 20, 1982

KEY PERSONNEL

TASK MANAGER
PATRICK SEARS X24216

Patrick Sears

MPA ANALYST
M. ANDERSON X27021

TASK REVIEWERS

NAME	BRANCH
T. CHENG	SEB/DL
S. CHAN, H. POLK	SEB/DE
L. REITER	GB/DE
#G. BAGCHI	EQB/DE

SCHEDULED COMPLETION

1978 ANNUAL REPORT	PHASE I – 1979 PHASE II – 1981
CURRENT	# 07-15-83

• PROBLEM DESCRIPTION

The seismic design process required by current NRC criteria includes the following sequence of events:

(a) Define the magnitude or intensity of the earthquake which will produce the maximum vibratory ground motion at the site (the safe shutdown earthquake or SSE).
(b) Determine the free-field ground motion at the site that would result if the SSE occurred.
(c) Determine the motion of site structures by modifying the free-field motion to account for the interaction of the site structures with the underlying foundation soil.
(d) Determine the motion of the plant equipment supported by the site structures.
(e) Compare the seismic loads, in appropriate combination with other loads, on structures, systems, and components important to safety, with the allowable loads.

While this seismic design sequence includes many conservative factors, certain aspects of the sequence may not be conservative for all plant sites. At present, it is believed that the overall sequence is adequately conservative. The objective of this program is to investigate selected areas of the seismic design sequence to determine their conservatism for all types of sites, to investigate alternate approaches to parts of the design sequence, to quantify the overall conservatism of the design sequence, and to modify the NRC criteria in the Standard Review Plan if changes are found to be justified. In this manner, this program will provide additional assurance that the health and safety of the public is protected, and if possible, reduce costly design conservatisms by improving (1) current seismic design requirements, (2) NRR's capability to quantitatively assess the overall adequacy of seismic design for nuclear plants in general.

• RES INTERFACE INFORMATION

Task 1.0 will provide input to the Qualification of Conservatisms in Present Seismic Methodology, one of the three objectives of the Seismic Research Program. Close coordination with RES will be maintained through meetings to exchange information.

• ACRS INTERFACE INFORMATION

The Extreme External Phenomena Subcommittee was briefed on the Phase I reports including the recommended changes to the Seismic Design Criteria.

• TECHNICAL ASSISTANCE CONTRACTS

Lawrence Livermore Laboratory (LLL), under contract to RES, reviewed all reports by 04-30-79. LLL report on recommendations for changes to the seismic design criteria was completed on 12-28-79. (NUREG/CR-1161).

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED

Contracts complete.

• POTENTIAL PROBLEMS

None.

• STATUS SUMMARY

All contractor work necessary to complete Phase I has been completed. Staff review of LLL report was completed on 10-30-80. A draft report presenting staff position has been prepared and is under review by the staff.

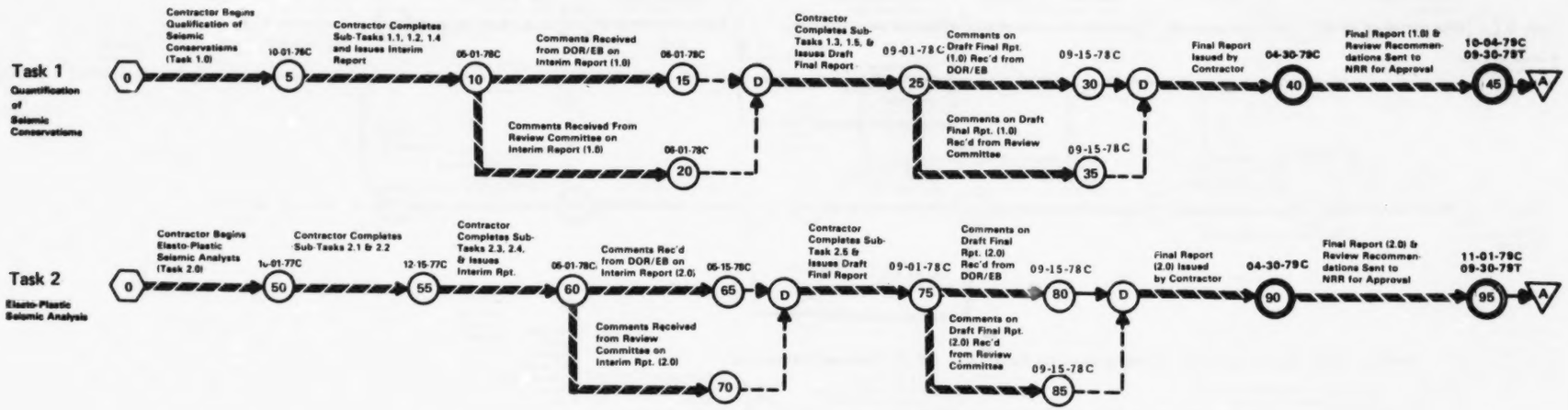
Schedule has been extended due to new requirement to incorporate CRGR implementation reviews.

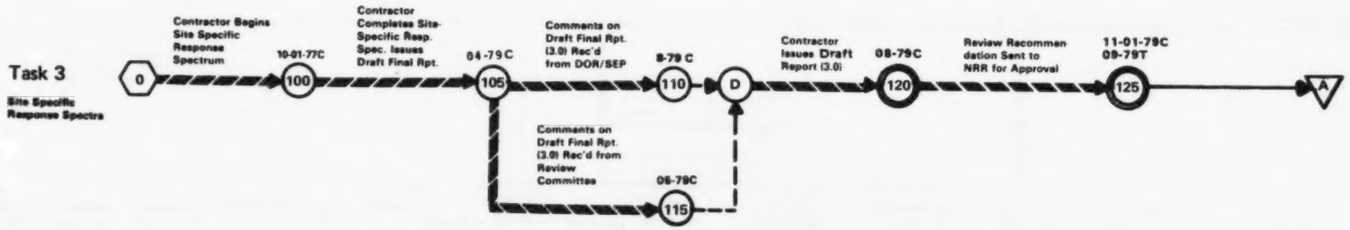
Draft manuscript has been completed and is being prepared for management review.

#INDICATES A CHANGE IN INFORMATION

SEISMIC DESIGN CRITERIA - SHORT TERM PROGRAM (A-40)

Phase 1

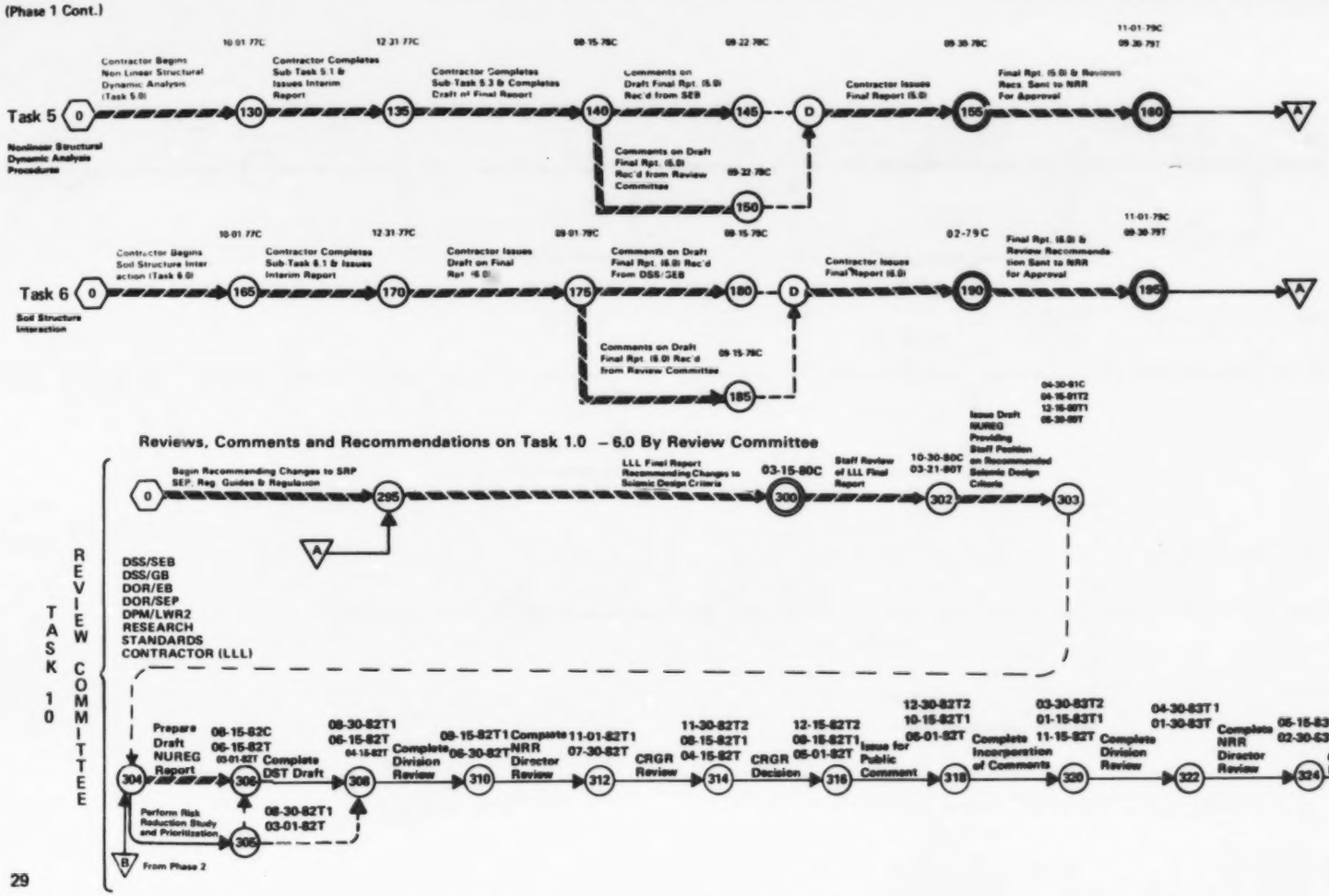

Task 1
Quantification of Seismic Conservatisms
Contractor Begins Qualification of Seismic Conservatisms (Task 1.0)
10-01-78C
Contractor Completes Sub-Tasks 1.1, 1.2, 1.4 and Issues Interim Report
05-01-78C
Comments Received from DOR/EB on Interim Report (1.0)
06-01-78C
Comments Received From Review Committee on Interim Report (1.0)
06-01-78C
Contractor Completes Sub-Tasks 1.3, 1.5, & Issues Draft Final Report
09-01-78C
Comments on Draft Final Rpt. (1.0) Rec'd from DOR/EB
09-15-78C
Comments on Draft Final Rpt. (1.0) Rec'd from Review Committee
09-15-78C
Final Report Issued by Contractor
04-30-79C
Final Report (1.0) & Review Recommendations Sent to NRR for Approval
10-04-79C
09-30-79T
Task 2
Elasto-Plastic Seismic Analysis
Contractor Begins Elasto-Plastic Seismic Analysis (Task 2.0)
10-01-77C
Contractor Completes Sub-Tasks 2.1 & 2.2
12-15-77C
Contractor Completes Sub-Tasks 2.3, 2.4, & Issues Interim Rpt.
06-01-78C
Comments Rec'd from DOR/EB on Interim Report (2.0)
06-15-78C
Comments Received from Review Committee on Interim Rpt. (2.0)
Contractor Completes Sub-Task 2.5 & Issues Draft Final Report
09-01-78C
Comments on Draft Final Rpt. (2.0) Rec'd from DOR/EB
09-15-78C
Comments on Draft Final Rpt. (2.0) Rec'd from Review Committee
09-15-78C
Final Report (2.0) Issued by Contractor
04-30-79C
Final Report (2.0) & Review Recommendations Sent to NRR for Approval
11-01-79C
09-30-79T



Task 4 Deleted

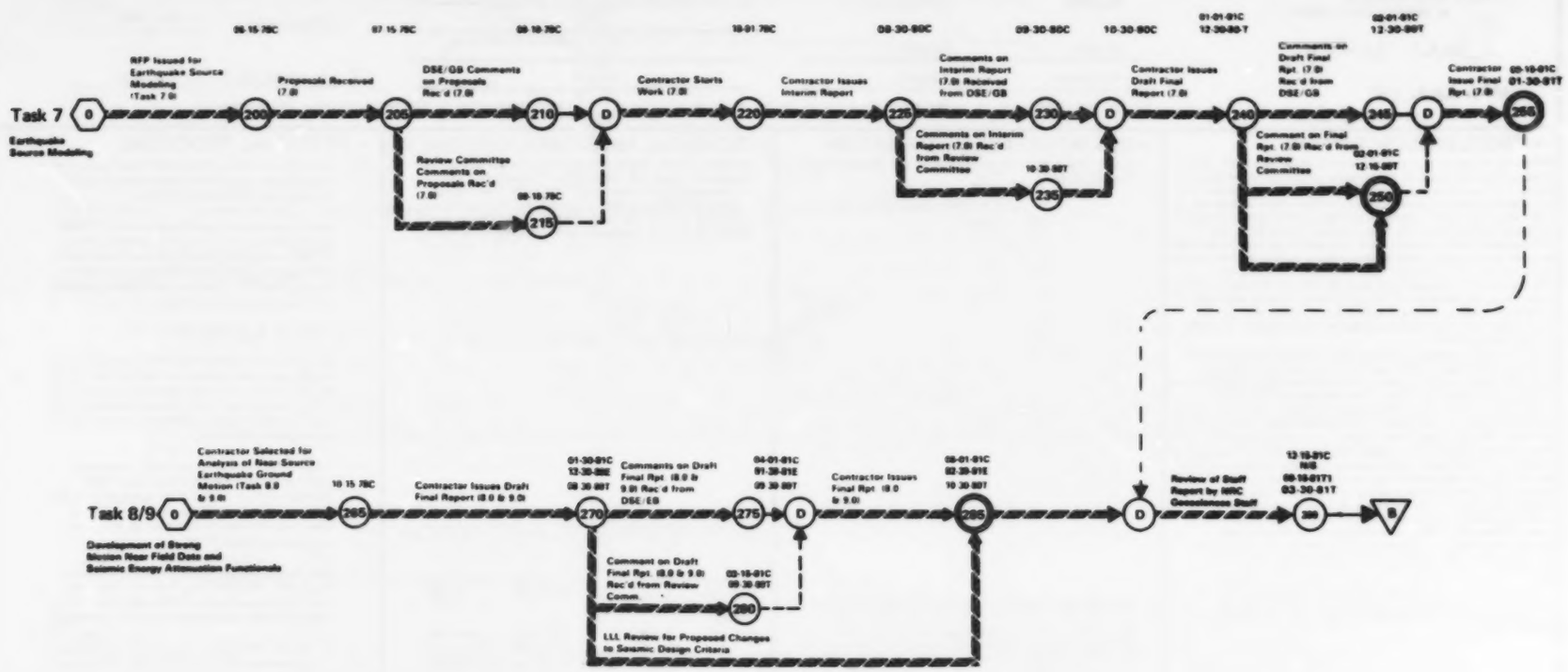
Seismic After Shocks

SEISMIC DESIGN CRITERIA - SHORT TERM PROGRAM (A-40)



Phase 2

SEISMIC DESIGN CRITERIA - SHORT TERM PROGRAM (A-40)



CONTAINMENT EMERGENCY SUMP PERFORMANCE (A-43)

AS OF WEEK ENDING AUGUST 20, 1982

KEY PERSONNEL

TASK MANAGER
A. SERKIZ (DST) X24217

MPA ANALYST
#M. ANDERSON X27021

TASK REVIEWERS

NAME	BRANCH
F. ORR	RSB/DSI
P. NORIAN	GIB/DST
W. BUTLER	CSB/DSI

SCHEDULED COMPLETION

ORIGINAL April 1982

CURRENT 07-15-83

• PROBLEM DESCRIPTION

Following a Loss of Coolant Accident (LOCA) in a PWR, water flowing from the break in the primary system would collect on the floor of containment. During the injection mode, water for core cooling and containment spray is drawn from a large supply tank. When the water reached a low level in the tank, pumps are realigned to draw from the containment. This is called the recirculation mode wherein water is drawn from the containment floor or sump and pumped to the primary system or containment spray headers. This program addresses the safety issue of adequate sump or suppression pool function in the recirculation mode. It is the objective of this program to develop improved criteria for design, testing, and evaluation which will provide better assurance that emergency sumps will function to satisfy system requirements.

The principal concerns are somewhat interrelated but are best discussed separately. One deals with the various kinds of insulation used on piping and components inside of containment. The concern being that break-initiated debris from the insulation could cause blockage of the sump or otherwise adversely affect the operation of the pumps, spray nozzles, and valves of the safety systems.

The second deals with the hydraulic performance of the sump as related to the hydraulic performance to safety systems supplied therefrom. Preoperational tests have been performed on a number of plants to demonstrate operability in the recirculation mode. Adverse flow conditions have been encountered requiring design and procedural modifications to eliminate them. These conditions, air entrainment, cavitation, and vortex formation, are aggrevated by blockage. If not avoided or suppressed, they could result in pump failure during the long term cooling phase following a LOCA.

The concerns relative to debris, blockage, and hydraulic performance also apply to boiling water reactors during recirculation from the suppression pools, and will also be addressed.

• RES INTERFACE INFORMATION

None; USI A-43 being managed by the Generic Issues Branch (GIB).

• ACRS INTERFACE INFORMATION

The ACRS staff is kept abreast of findings and developments on an informal basis. A briefing on the current findings and status of resolution was held December 3, 1981.

• TECHNICAL ASSISTANCE CONTRACTS

FIN No. A1237, "Containment Emergency Sump Performance", and FIN No. A1296, "Technical Assistance for Resolution of USI A-43", are being funded by RES and NRR respectively. This work is managed by the GIB Task Manager and these combined efforts are expected to be concluded in FY82.

FIN. NO.	CONTRACTOR		OBLIGATED	EXPENDED
FY 81:				
A1237	Sandia		$225,000	$101,000
A1296	Sandia		$120,000	$ 75,100
FY 82:				
# A1237	Sandia	Qtr 1	$334,000	$103,000
		Qtr 2	$315,000	$125,000
		Qtr 3	$315,000	$245,000
		Qtr 4	–	–
# A1296	Sandia	Qtr 1	$ 71,900	$ 38,000
		Qtr 2	$194,900	$119,000
		Qtr 3	$344,900	$128,000
		Qtr 4	–	–

• POTENTIAL PROBLEMS

#1) Completion of risk analysis and development of value/impact evaluations are the pacing items for preparing the CRGR package.

#2) Recent redirection regarding contents of resolution NUREGs and in required contents of the CRGR package has resulted in some uncertainty with respect to meeting near-term (next 2-3 months) schedule. Meeting milestones noted for A-43 assumes first round success.

• STATUS SUMMARY

#1) NRR divisional concurrance has been obtained for the technical findings set forth in the "draft" NUREG-0897, "Containment Emergency Sump Performance," July 1982. These findings will be used to prepare recommendations to the CRGR for resolving USI A-43.

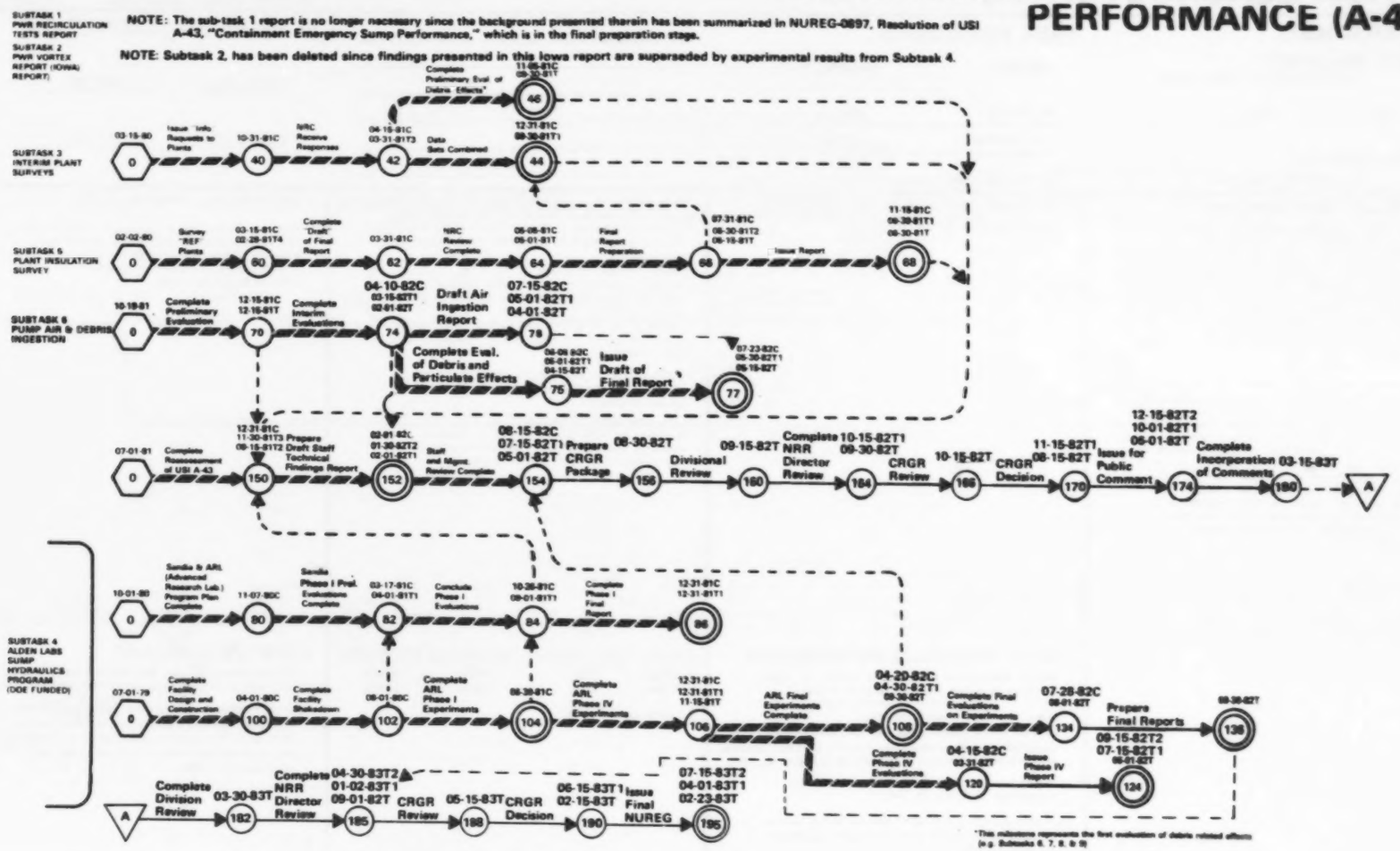
#2) Preparation of a CRGR package is underway utilizing the technical findings from "draft" NUREG-0897, a probabilistic assessment of the risks associated with failure of the sump and costs attendant to implementation of recommendations to the CRGR.

#3) Concluding evaluations of outstanding reports from ARL, Burns & Roe, Inc., and CREARE, Inc. are expected to be completed in August 1982. Recently published reports are as follows:

- NUREG/CR-2758, "A Parametric Study of Containment Emergency Sump Performance" (07/82)
- NUREG/CR-2760, "Assessment of Scale Effects on Vortexing, Swirl, and Inlet Losses in Large Scale Sump Models" (06/82)
- NUREG/CR-2772, "Hydraulic Performance of Pump Suction Inlets for Emergency Core Cooling Systems in Boiling Water Reactors" (06/82)

# INDICATES A CHANGE IN INFORMATION

CONTAINMENT EMERGENCY SUMP PERFORMANCE (A-43)


SUBTASK 1
PWR RECIRCULATION TESTS REPORT
NOTE: The sub-task 1 report is no longer necessary since the background presented therein has been summarized in NUREG-0897, Resolution of USI A-43, "Containment Emergency Sump Performance," which is in the final preparation stage.
SUBTASK 2
PWR VORTEX REPORT (IOWA REPORT)
NOTE: Subtask 2, has been deleted since findings presented in this Iowa report are superseded by experimental results from Subtask 4.
SUBTASK 3
INTERIM PLANT SURVEYS
SUBTASK 5
PLANT INSULATION SURVEY
SUBTASK 6
PUMP AIR & DEBRIS INGESTION
SUBTASK 4
ALDEN LABS SUMP HYDRAULICS PROGRAM (DOE FUNDED)
Draft Air Ingestion Report
Complete Eval. of Debris and Particulate Effects
Issue Draft of Final Report
Prepare CRGR Package
Divisional Review
CRGR Review
CRGR Decision
Issue Final NUREG
Issue for Public Comment
Complete Incorporation of Comments
Prepare Final Reports
Issue Phase IV Report
Complete Phase IV Evaluations
ARL Final Experiments Complete
Complete Final Evaluations on Experiments
*This milestone represents the first evaluation of debris related effects (e.g. Subtasks 6, 7, 8, & 9)

STATION BLACKOUT (A-44)

AS OF WEEK ENDING AUGUST 20, 1982

KEY PERSONNEL

TASK MANAGER
P. BARANOWSKY (RES) X38980

P. Nowa. for P Baranowsky

MPA ANALYST
L. LANIER X28102

TASK REVIEWERS

NAME	BRANCH
#L. ENGLE	ORB/DL
O. CHOPRA	PSB/DSI

SCHEDULED COMPLETION

ORIGINAL JUNE 1982

CURRENT 10-15-83

• PROBLEM DESCRIPTION

Electric power for safety systems at nuclear power plants is supplied by two redundant and independent divisions. Each of these electrical divisons includes an offsite alternating current (A.C.) source, an onsite A.C. source (usually diesel-generators), and a direct current (D.C.) source. Appendix A to 10 CFR 50 defines a total loss of offsite power as an anticipated occurrence, and as such, it is required that an independent emergency onsite power supply be provided at nuclear power plants.

The unlikely, but possible loss of A.C. power (that is, the loss of A.C. power from the offsite source and from the onsite source) is referred to as a station blackout. In the event of a station blackout, the capability to cool the reactor core would be dependent on the availability of systems which do not require A.C. power supplies, and on the ability to restore A.C. power in a timely manner. The concern is that the occurrence of a station blackout may be a relatively high probability event and that the consequences of this event may be unacceptable, for example, severe core damage may result.

• RES INTERFACE INFORMATION

RES is managing the Task for NRR

• ACRS INTERFACE INFORMATION

Station Blackout is related to a number of ACRS concerns regarding the reliability of power systems. This task will be coordinated with the committee as the task progresses.

Task action plan outline was discussed with AC/DC Power Reliability Subcommittee on January 22, 1981. Another subcommittee briefing was held on October 30, 1981.

A program status briefing was given to this subcommittee on March 30, 1982.

#A presentation to the ACRS subcommittee summarizing the results of the contractor work is planned for Sept. 8, 1982.

• TECHNICAL ASSISTANCE CONTRACTS

#Technical Assistance in the amount of $50K is estimated for this task. The following elements are included in this program:

#ORNL FIN B0744 $800K –
Evaluate expected frequency and duration of offsite (preferred) power losses at nuclear power plants.
Estimate the reliability and evaluate the dominant factors affecting the reliability of emergency A.C. power supplies.

#$50K in funding was recently added to perform statistical correlation and trend analysis of diesel generator data.

SNL FIN A1302 $250K –
Evaluate the risks posed by station blackout accidents and assess the effectiveness of safety improvements in reducing those risks.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED*
# B0744	ORNL	$800K	$580K
A1302	SNL	$250K	$240K

*As of June 30, 1982

• POTENTIAL PROBLEMS

Results of the extensive contractor work indicate that because of availability in the systems design at different plants, implementation will need to be on a plant specific or a plant group specific basis in order to be cost effective.

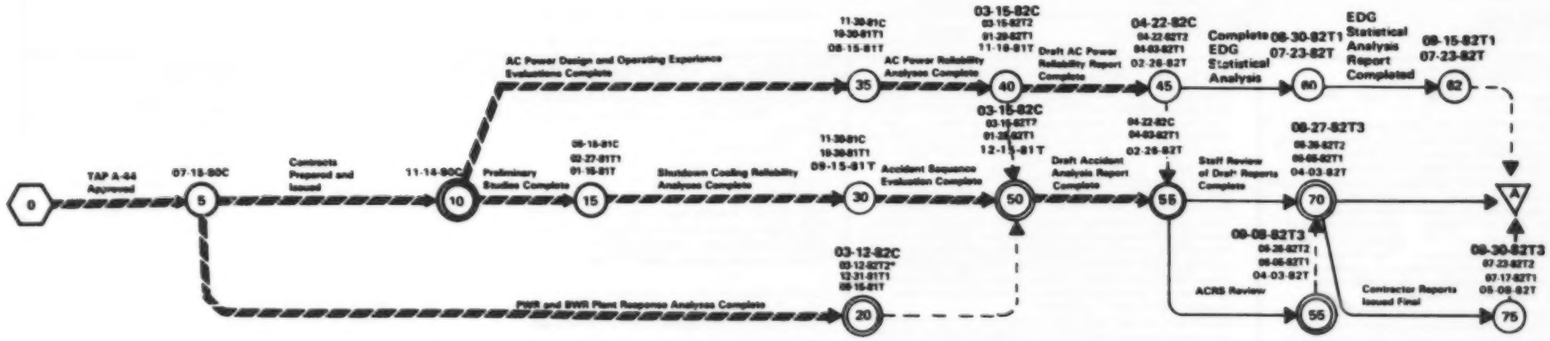
• STATUS SUMMARY

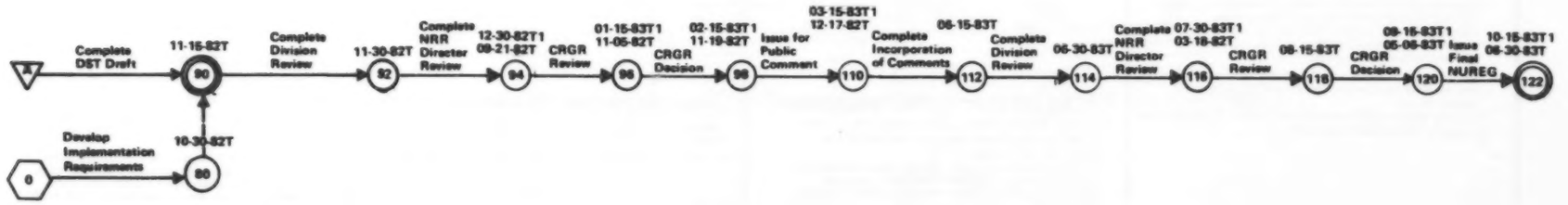
A Task Action Plan for this task was issued in July 1980 and work is proceeding in accordance with this plan.

#Draft contractor reports from Sandia and ORNL have been received and are being reviewed by the NRC staff. A contractor presentation summarizing the results of the studies was made to the staff on August 11, 1982.

# INDICATES A CHANGE IN INFORMATION

STATION BLACKOUT (A-44)





*Preliminary analyses completed 9-81.

SHUTDOWN DECAY HEAT REMOVAL REQUIREMENTS (A-45)

AS OF WEEK ENDING AUGUST 20, 1982

KEY PERSONNEL

TASK MANAGER
ANDREW MAPCHESE X24712

MPA ANALYST
B. ST. MARY X28330

TASK REVIEWERS

NAME	BRANCH
T. MARSH	RSB/DSI
F. ROSA	ICSB/DSI
M. SRINIVASAN	PSB/DSI
E. McPEEK	SSPB/DL
#P. WAGNER	ORB-4/DL
M. CUNNINGHAM	DRA/RES

SCHEDULED COMPLETION

ORIGINAL	10/85 PRELIMINARY NETWORK
CURRENT	#10-30-85

• PROBLEM DESCRIPTION

Task A-45 was approved as a USI by the NRC in December 1980.

Although many improvements to the steam generator auxiliary feedwater system were required of the reactor manufacturers by the NRC following the TMI-2 accident, the staff feels that providing an alternative means of decay heat removal could substantially increase the plants' capability to deal with a broader spectrum of transients and accidents and potentially could, therefore, significantly reduce the overall risk to the public. Consequently, Task A-45 will investigate alternative means of decay heat removal in PWR plants, including but not limited to using existing equipment where possible. This Unresolved Safety Issue will also investigate the need and possible design requirements for improving reliability of decay heat removal systems in boiling water reactors (BWRs).

The overall purpose of Task A-45 is to evaluate the adequacy of current licensing design requirements, in order to ensure that nuclear power plants do not pose an unacceptable risk due to failure to remove shut-down decay heat. The objective will be to develop a comprehensive and consistent set of shutdown cooling requirements for existing and future LWRs, including the study of alternative means of shutdown decay heat removal and of diverse "dedicated" systems for this purpose.

The approach taken to this problem comprises the following subtasks:

- Development of criteria to judge acceptability of decay heat removal systems in existing & future plants.
- Development of means for improvement of DHR systems.
- Assessment of existing plants to identify those in which DHR systems require improvement.

Development of a plan for implementing new requirements, if any, for DHR systems required to meet the acceptance criteria above.

The interrelation and relative timing of each of the sub-tasks is shown on the schedule network.

• RES INTERFACE INFORMATION

Close coordination and cooperation will be required on Task A-45 between NRR and RES. RES assistance will be required from the Divisions of Risk Analysis, Accident Evaluation, and Engineering Technology. The Division of Risk Analysis will provide technical input from their Sandia Laboratory Program on Alternate Decay Heat Removal Concepts, technical evaluations relative to reliability and risk assessment for shutdown decay heat removal systems, and input from Task A-44, "Station Blackout," relative to shutdown cooling systems. The Division of Accident Evaluation will provide technical input relative to the transient response of existing and improved shutdown decay heat removal systems to transient events and small LOCAs. This will also include performing (in-house, contractors) detailed thermal-hydraulics analyses where required to support existing and improved decay heat removal systems behavior under transient and accident conditions.

The Division of Engineering Technology will provide assistance in the preparation and publication (i.e., Rules, Regulatory Guides, Standard Review Plans) of a consistent and comprehensive set of shutdown decay heat removal requirements. At the outset, particular attention will need to be focused on the further development of Regulatory Guide 1.139, "Guidance for Residual Heat Removal to Achieve and Maintain Cold Shutdown," Revision 1.

• ACRS INTERFACE INFORMATION

#– Task Manager briefed the full committee on July 9, 1982 on the present plans and content of the A-45 Task Action Plan (TAP).

– ACRS Subcommittee on Decay Heat Removal Systems (DHRS) met on March 16, 1982, and Task Manager provided the subcommittee with an update of the TAP.

– Further meetings with the full committee and subcommittee on DHRS will be held as the work on A-45 progresses and certain pre-determined milestones are completed.

• TECHNICAL ASSISTANCE CONTRACTS

#– Supplemented UCLA Contract (FIN-B6835) on May 1, 1981 by 10K to provide direct technical assistance to Task Manager to develop a Task Action Plan for resolution of A-45.

#– Implemented a technical assistance contract on August 5, 1981 with BNL (FIN-A-3381) for $100K to make a determination whether it is feasible to separate all currently operating U.S. commercial LWR's into groups based on whether they have similar DHRS design characteristics to the plants covered in WASH-1400, RSSMAP, and IREP.

#Implemented a technical assistance contract on May 10, 1982 with Sandia (FIN A1309) to provide overall project management, technical direction and integration for the entire Task A-45 program, including selection and management of subcontractors if needed.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
B6835	UCLA	$10,000	$10,000
# A3381	BNL	$100.000	$72,000
#A1309	Sandia	$400.000	$98,000

• POTENTIAL PROBLEMS

Potential problem areas identified to date are:

- Obtaining sufficient NRR manpower to work on Task A-45.
- Development of appropriate reliability or quantitative goals for Task A-45 and translation of probabilistic results into licensing requirements.
- Program funding must be approved and obtained.
- Obtaining necessary operating experience on shutdown cooling systems including the most current DHR system design information resulting from Post-TMI changes.
- Uncertainty in the quality of information that will be available from ongoing reliability and risk assessments, on what schedule, and whether information can be extrapolated to all operating plants.
- Close coordination and cooperation needed within NRR and between NRR and RES.
- Number of plants that need to be assessed may be significantly greater than the plants that will have a PRA or reliability study performed.
- Time to obtain approval of funding by NRR and the Senior Contract Review Board has been greater than expected by about six months (See Status Summary).

• STATUS SUMMARY

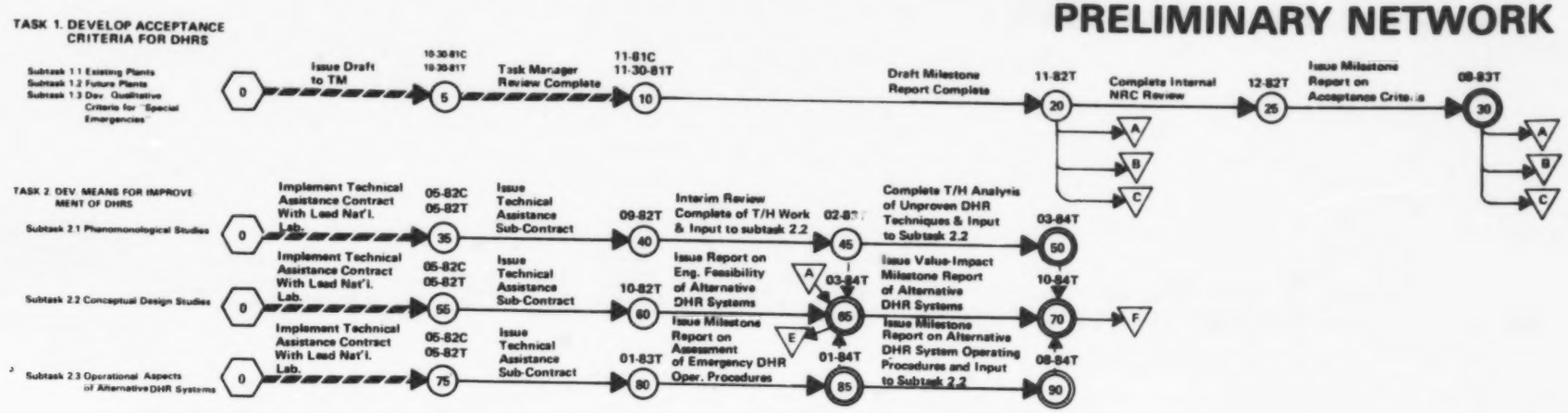
The task manager has been selected and a draft Task Action Plan was published on May 22, 1981.

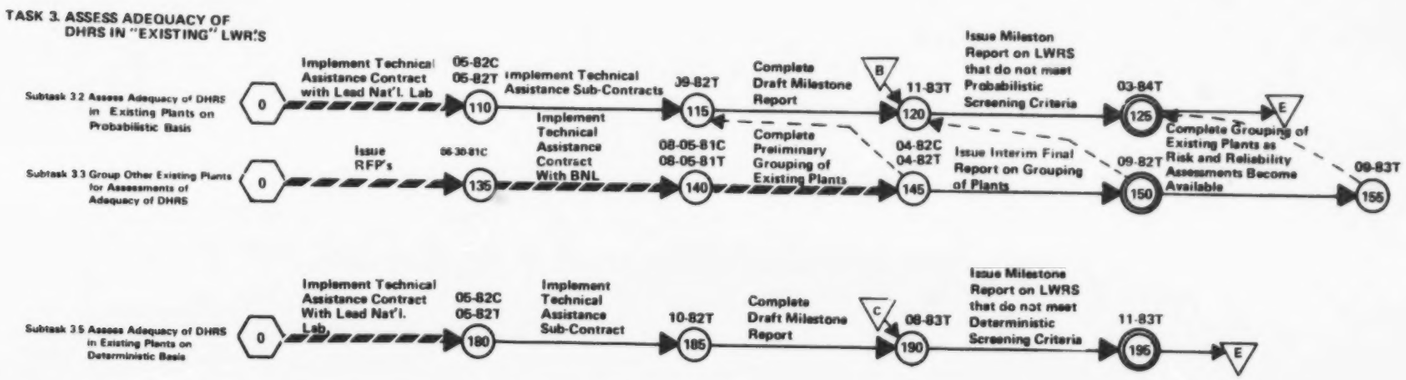
Based on internal staff and ACRS comments, Revision 0 of TAP A-45 was approved on October 7, 1981.

The Director of NRR did not approve the requested funding level for Task A-45 technical assistance work, and a "HOLD" was placed on the program in Nov. 1981. Various options were considered for Task A-45, most of which included reduced scope, funding and completion time. A recommended option was approved by The Director of NRR in Feb. 1982. A technical assistance contract package was approved by the Senior Contract Review Board on April 9, 1982 and implemented with the #Contractor (Sandia) on May 10, 1982. Revision 1 of TAP A-45 consistent with the above actions was issued on June 2, 1982.

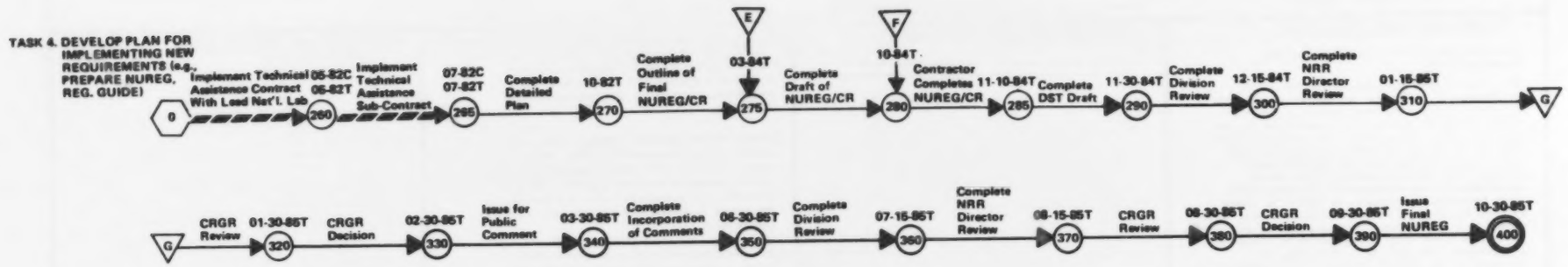
# INDICATES A CHANGE IN INFORMATION

SHUTDOWN DECAY HEAT REMOVAL REQUIREMENTS (A-45) PRELIMINARY NETWORK


TASK 1. DEVELOP ACCEPTANCE CRITERIA FOR DHRS
Subtask 1.1 Existing Plants
Subtask 1.2 Future Plants
Subtask 1.3 Dev. Qualitative Criteria for "Special Emergencies"
Issue Draft to TM
10-30-81C
10-30-81T
Task Manager Review Complete
11-81C
11-30-81T
Draft Milestone Report Complete
11-82T
Complete Internal NRC Review
12-82T
Issue Milestone Report on Acceptance Criteria
09-83T
TASK 2. DEV. MEANS FOR IMPROVEMENT OF DHRS
Subtask 2.1 Phenomonological Studies
Implement Technical Assistance Contract With Lead Nat'l. Lab.
05-82C
05-82T
Issue Technical Assistance Sub-Contract
09-82T
Interim Review Complete of T/H Work & Input to subtask 2.2
02-83T
Complete T/H Analysis of Unproven DHR Techniques & Input to Subtask 2.2
03-84T
Subtask 2.2 Conceptual Design Studies
Implement Technical Assistance Contract With Lead Nat'l. Lab.
05-82C
05-82T
Issue Technical Assistance Sub-Contract
10-82T
Issue Report on Eng. Feasibility of Alternative DHR Systems
03-84T
Issue Value-Impact Milestone Report of Alternative DHR Systems
10-84T
Subtask 2.3 Operational Aspects of Alternative DHR Systems
Implement Technical Assistance Contract With Lead Nat'l. Lab.
05-82C
05-82T
Issue Technical Assistance Sub-Contract
01-83T
Issue Milestone Report on Assesment of Emergency DHR Oper. Procedures
01-84T
Issue Milestone Report on Alternative DHR System Operating Procedures and Input to Subtask 2.2
08-84T


TASK 3. ASSESS ADEQUACY OF DHRS IN "EXISTING" LWR'S
Subtask 3.2 Assess Adequacy of DHRS in Existing Plants on Probabilistic Basis
Implement Technical Assistance Contract with Lead Nat'l. Lab
05-82C
05-82T
Implement Technical Assistance Sub-Contracts
09-82T
Complete Draft Milestone Report
11-83T
Issue Milestone Report on LWRS that do not meet Probabilistic Screening Criteria
03-84T
Subtask 3.3 Group Other Existing Plants for Assessments of Adequacy of DHRS
Issue RFP's
06-30-81C
Implement Technical Assistance Contract With BNL
08-05-81C
08-05-81T
Complete Preliminary Grouping of Existing Plants
04-82C
04-82T
Issue Interim Final Report on Grouping of Plants
09-82T
Complete Grouping of Existing Plants as Risk and Reliability Assessments Become Available
09-83T
Subtask 3.5 Assess Adequacy of DHRS in Existing Plants on Deterministic Basis
Implement Technical Assistance Contract With Lead Nat'l. Lab.
05-82C
05-82T
Implement Technical Assistance Sub-Contract
10-82T
Complete Draft Milestone Report
08-83T
Issue Milestone Report on LWRS that do not meet Deterministic Screening Criteria
11-83T

SHUTDOWN DECAY HEAT REMOVAL REQUIREMENTS (A-45) (CONTINUED)
PRELIMINARY NETWORK



SEISMIC QUALIFICATION OF EQUIPMENT IN OPERATING PLANTS (A-46)

AS OF WEEK ENDING AUGUST 20, 1982

KEY PERSONNEL

TASK MANAGER
PATRICK SEARS X24216

Patrick Sears

MPA ANALYST
B. ST. MARY X28330

TASK REVIEWERS

NAME	BRANCH
JERRY JACKSON	EQB/DE

SCHEDULED COMPLETION

ORIGINAL 12-15-83

CURRENT #06-01-84

• PROBLEM DESCRIPTION

Task A-46 was approved as a USI by the NRC December, 1980.

The design criteria and methods for the seismic qualification of mechanical and electrical equipment in nuclear power plants have undergone significant change during the course of the commercial nuclear power program. Consequently, the margins of safety provided in existing equipment to resist seismically induced loads and perform the intended safety functions may vary considerably. The seismic qualification of the equipment in operating plants must, therefore, be reassessed to ensure the ability to bring the plant to a safe shutdown condition when subject to a seismic event. The objective of this Unresolved Safety Issue is to establish an explicit set of guidelines that could be used to judge the adequacy of the seismic qualification of mechanical and electrical equipment at all operating plants in lieu of attempting to backfit current design criteria for new plants. This guidance will concern equipment required to safely shutdown the plant, as well as equipment whose function is not required for safe shutdown, but whose failure could result in adverse conditions which might impair shutdown functions. Also, explicit guidelines will be established for use in requalifying equipment whose seismic qualification was found to be inadequate.

A preliminary breakdown of the tasks is as follows:

Task 1	Develop Basis for Equipment to be Qualified
Task 2	Survey of Equipment Qualification Methods
Task 3	Develop Methods for In-Situ Qualification of Equipment in Operating Plants
Task 4	Seismic Qualification of Equipment Using Non-Nuclear Plant Dynamic Response Information
Task 5	Develop Guidelines for Qualification Response Spectra
Task 6	Document Results of Task A-46 and Prepare a Final NUREG

• RES INTERFACE INFORMATION

RES has placed a contract with Southwest Research Institute to survey seismic qualification methodology.

• ACRS INTERFACE INFORMATION

A preliminary Task Action Plan was presented to the ACRS. The staff plans to meet with the ACRS at appropriate times to present the status of program results.

• TECHNICAL ASSISTANCE CONTRACTS

#EG&G is well into Task 3. They have completed 90% of their library search and contacted test laboratories concerning in-situ qualification of equipment.

#A plant owners group has collected a significant amount of data on equipment in non-nuclear plants which have undergone earthquakes. Similar data on equipment in nuclear plants has also been collected. LLL, as a part of Task 4, is evaluating the feasibility of using these data for equipment qualification.

#BNL has completed work on developing methodology to select equipment to be qualified on a typical PWR. Work on a typical BWR has started.

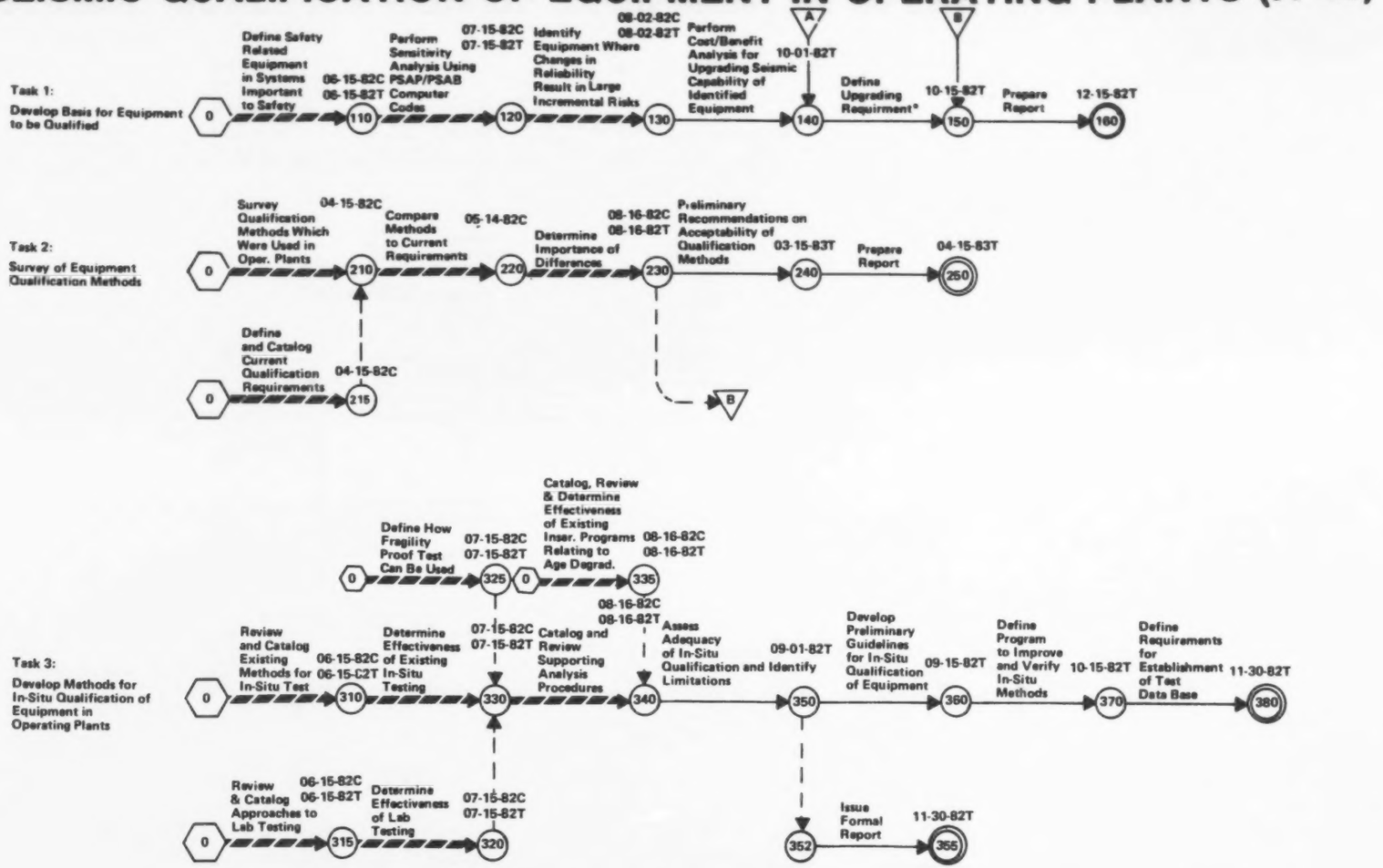
FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
#A0423	LLNL	$75K	$50K (est.)
#A6474	INEL	$165K	$120K (est.)
#A3397	BNL	$126K	$100K (est.)

• POTENTIAL PROBLEMS

• STATUS SUMMARY

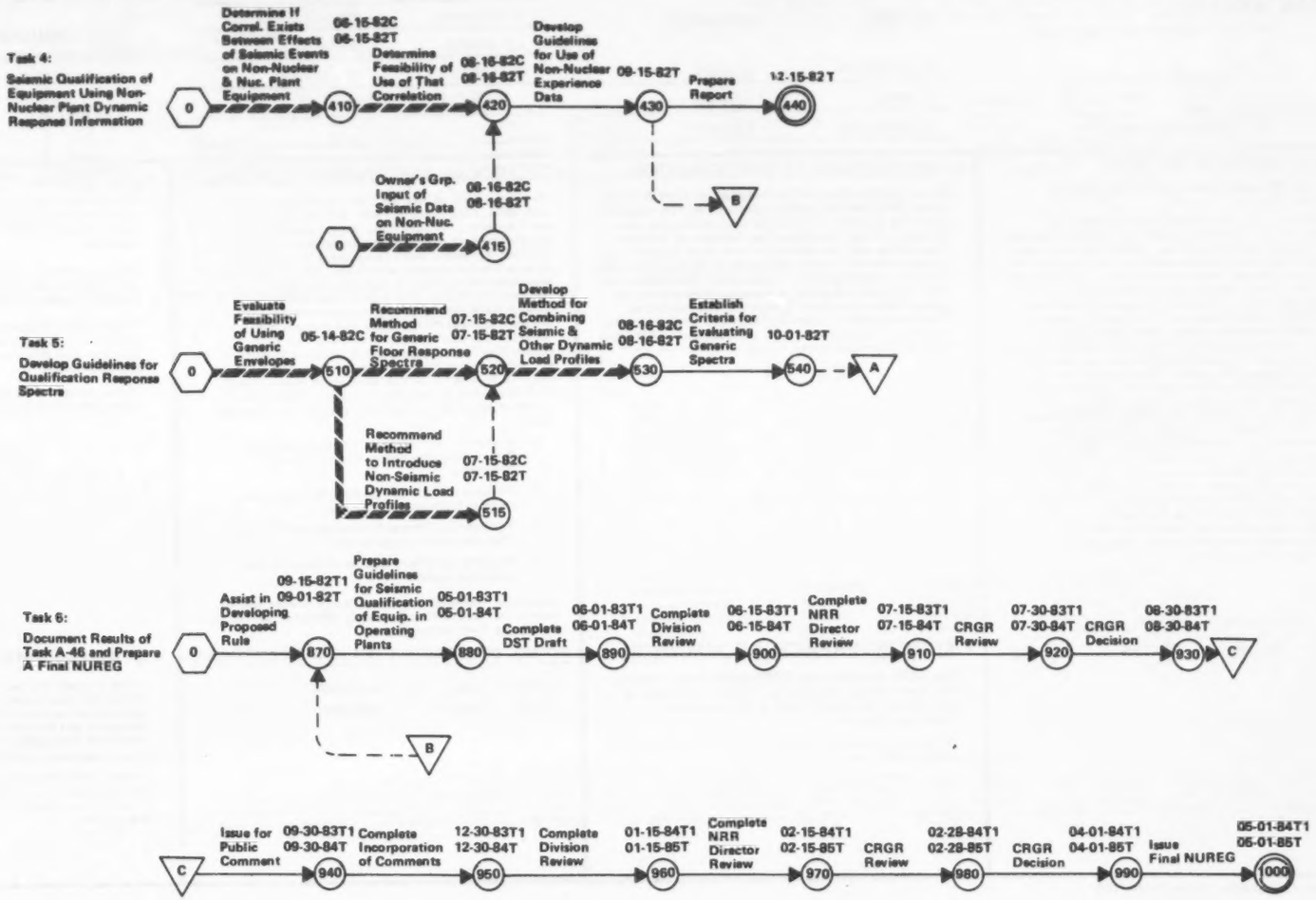
#Work is progressing on all tasks with all contractors well into their subtasks. Meetings have taken place with members of the Owner's Group who are surveying nuclear and non-nuclear equipment, and LLNL is evaluating the feasibility of using experience data to qualify equipment.

SEISMIC QUALIFICATION OF EQUIPMENT IN OPERATING PLANTS (A-46)



*Required Upgrading: (1) Upgrade None, (2) Upgrade High Risk Only, (3) Upgrade All.

SEISMIC QUALIFICATION OF EQUIPMENT IN OPERATING PLANTS (A-46) Continued



SAFETY IMPLICATIONS OF CONTROL SYSTEMS (A-47)

AS OF WEEK ENDING AUGUST 20, 1982

KEY PERSONNEL

TASK MANAGER
ANDREW SZUKIEWICZ (GIB) X24713

MPA ANALYST
M. ANDERSON X27021

TASK REVIEWERS

NAME	BRANCH
D. BASDEKAS	DFO/RES
C. ROSSI	ICSB/DSI
S. NEWBERRY	RRAB/DST
F. ORR	RSB/DSI
M. CHIRAMAL	PSU/AEOD
J. T. BEARD	ORAB/DL
W. KENNEDY	PTRB/DHFS

SCHEDULED COMPLETION

1978 ANNUAL REPORT	NOT DETERMINED
CURRENT	N/S

• PROBLEM DESCRIPTION

Task A-47 was approved as a USI by the NRC in December 1980.

This issue concerns the potential for accidents or transients being made more severe as a result of control system failures or malfunctions. These failures or malfunctions may occur independently or as a result of the accident or transient under consideration and would be in addition to any control system failure that may have initiated the event. Although it is generally believed that control system failures are not likely to result in loss of safety functions which could lead to serious events or result in conditions that safety systems are not able to cope with, in-depth studies have not been performed to support this belief. The potential for an accident that would affect a particular control system - and the effects of the control system failures - will differ from plant to plant. Therefore, it is not likely that it will be possible to develop generic answers to these concerns, but rather plant-specific reviews will be required. The purpose of this Unresolved Safety Issue is to define generic criteria that may be used for plant-specific reviews. A specific subtask of this issue will be to study the steam generator overfill transient in PWRs and the reactor overfill transient in BWRs to determine and define the need for preventive and/or mitigating design measures to accommodate this transient.

• RES INTERFACE INFORMATION

Close coordination will be required on Task A-47 between NRR and RES. RES assistance will be required from the Division of Facilities Operations and the Division of Risk Analysis. The Division of Facilities Operations will provide input from their research programs (currently being developed in conjunction with A-47) on control system interactions. The Division of Risk Analysis will provide input on their current ongoing ORNL program ("Evaluation of Pressurized Thermal Shock"). Other research activities are being evaluated to determine their significance to A-47 and may be identified later.

• ACRS INTERFACE INFORMATION

The ACRS Subcommittee on Electrical Power Systems (Dr. Kerr) met on the issue on 1/23/81 to initiate an approximate six month study at the request of Chairman Ahearne.

Elements of conceptual Task Action Plan were discussed with ACRS on 06-07-81.

• TECHNICAL ASSISTANCE CONTRACTS

During the week of July 20, 1981, meetings were held with INEL, EG&G (Idaho Falls), Battelle Northwest and Lawrence Livermore Labs. for the purpose of discussing the A-47 objectives and to assess the type of activities that could best be conducted by these organizations. Similar meetings with ORNL and Sandia Labs. were held July 9, 1981. A Technical Assistance (T.A.) contract with ORNL (FIN B-0467) was established through NRR/RES. RES will conduct a review of two different types of PWR designs, and perform the activities identified in Tasks 1, 2, 3 and 4 of Task Action Plan A-47. A separate T.A. contract to perform the review on one BWR type design was established with INEL (FIN A-6477). A T.A. contract to review a third PWR design will be established in late FY1982. The technical assistance contracts will perform the following tasks:

1. Evaluate Control System Failures Leading to Steam Generator and/or Reactor Overfill Transients
2. Evaluate Control System Failures Leading to Reactor Overcooling Transients
3. Evaluate (All Others) Control System Actions That Have Safety Implications
4. Evaluate the Effects of Loss of Power to the Control Systems

Additional Technical Assistance Contract work will be needed for Task A-47 and is being defined as part of the developing Task Action Plan.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
A-6477	INEL	$ 50,000	
B-0467	ORNL	$600,000	

• POTENTIAL PROBLEMS

Obtain necessary operating experience information on Non-Safety Control Systems for the Nuclear Steam System (NSS) and the Balance of Plant (BOP) systems.

Obtain necessary design information (i.e., schematics and systems descriptions) for the Non-Safety Control System designs for the NSS and the BOP designs.

Traditionally, licensees were not required to provide design or operating experience on Non-Safety Systems and therefore this information may be difficult to obtain.

Obtain timely concurrence and approval of Task Action Plan A-47 by ICSB/DSI/NRR.

• STATUS SUMMARY

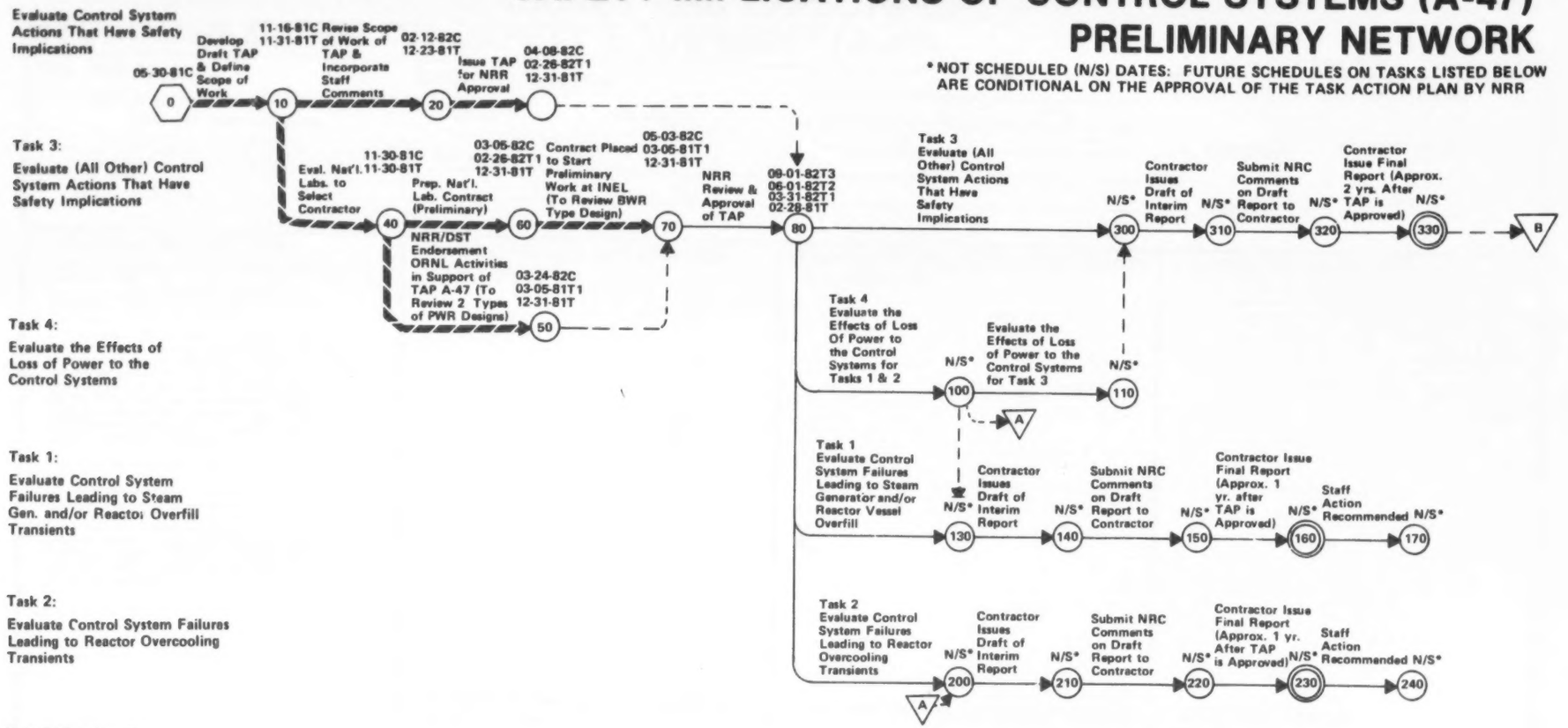
A task manager has been selected. A revised Draft Task Action Plan was issued in April, 1982 for staff concurrence and approval. A conceptual network has been developed and the task work schedules are being developed. Management approval of the Task Action
#Plan is expected by 09/01/82.

A work order was placed with INEL (May 3, 1982) to start preliminary work. Model development for computer simulation of the control systems is underway at ORNL.

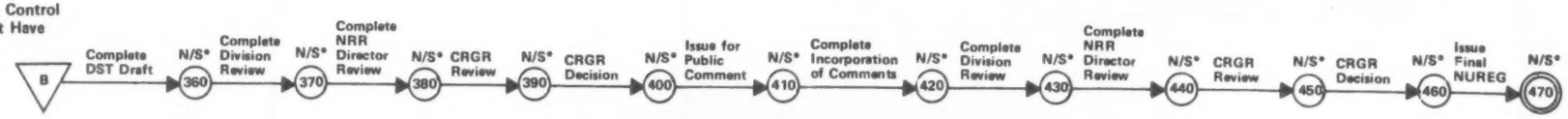
# INDICATES A CHANGE IN INFORMATION

SAFETY IMPLICATIONS OF CONTROL SYSTEMS (A-47)
PRELIMINARY NETWORK

* NOT SCHEDULED (N/S) DATES: FUTURE SCHEDULES ON TASKS LISTED BELOW ARE CONDITIONAL ON THE APPROVAL OF THE TASK ACTION PLAN BY NRR


Evaluate Control System Actions That Have Safety Implications
05-30-81C
Develop Draft TAP & Define Scope of Work
11-16-81C Revise Scope 11-31-81T of Work of TAP & Incorporate Staff Comments
02-12-82C 12-23-81T
Issue TAP for NRR Approval
04-08-82C 02-26-82T1 12-31-81T
0
10
20
Task 3: Evaluate (All Other) Control System Actions That Have Safety Implications
11-30-81C Eval. Nat'l. 11-30-81T Labs. to Select Contractor
Prep. Nat'l. Lab. Contract (Preliminary)
03-06-82C 02-26-82T1 12-31-81T
Contract Placed to Start Preliminary Work at INEL (To Review BWR Type Design)
05-03-82C 03-05-81T1 12-31-81T
NRR Review & Approval of TAP
09-01-82T3 06-01-82T2 03-31-82T1 02-26-81T
40
60
70
80
Task 4: Evaluate the Effects of Loss of Power to the Control Systems
NRR/DST Endorsement ORNL Activities in Support of TAP A-47 (To Review 2 Types of PWR Designs)
03-24-82C 03-05-81T1 12-31-81T
50
Task 3 Evaluate (All Other) Control System Actions That Have Safety Implications
N/S*
300
Contractor Issues Draft of Interim Report
310
Submit NRC Comments on Draft Report to Contractor
320
Contractor Issue Final Report (Approx. 2 yrs. After TAP is Approved)
330
B
Task 4 Evaluate the Effects of Loss Of Power to the Control Systems for Tasks 1 & 2
100
Evaluate the Effects of Loss of Power to the Control Systems for Task 3
110
A
Task 1: Evaluate Control System Failures Leading to Steam Generator and/or Reactor Vessel Overfill
130
Contractor Issues Draft of Interim Report
140
Submit NRC Comments on Draft Report to Contractor
150
Contractor Issue Final Report (Approx. 1 yr. after TAP is Approved)
160
Staff Action Recommended
170
Task 2: Evaluate Control System Failures Leading to Reactor Overcooling Transients
200
Contractor Issues Draft of Interim Report
210
Submit NRC Comments on Draft Report to Contractor
220
Contractor Issue Final Report (Approx. 1 yr. After TAP is Approved)
230
Staff Action Recommended
240
A
Task 1: Evaluate Control System Failures Leading to Steam Gen. and/or Reactor Overfill Transients
Task 2: Evaluate Control System Failures Leading to Reactor Overcooling Transients

Task 3 (Continued): Evaluate (All Other) Control System Actions That Have Safety Implications



HYDROGEN CONTROL MEASURES AND EFFECTS OF HYDROGEN BURNS ON SAFETY EQUIPMENT (A-48)

KEY PERSONNEL

TASK MANAGER
CLIFFORD ANDERSON X27486

Clifford J. Anderson 8/30/82

MPA ANALYST
B. ST. MARY X28330

TASK REVIEWERS

NAME	BRANCH
NOT ASSIGNED	

AS OF WEEK ENDING AUGUST 20, 1982

SCHEDULED COMPLETION

ORIGINAL: NOT DETERMINED

CURRENT: N/S

• PROBLEM DESCRIPTION

Task A-48 was approved as a USI by the NRC in December 1980.

Postulated reactor accidents which result in a degraded or melted core can result in generation and release to the containment of large quantities of hydrogen. The hydrogen is formed from the reaction of the zirconium fuel cladding with steam at high temperatures and/or by radiolysis of water. Experience gained from the TMI-2 accident indicates that we may want to require more specific design provisions for handling larger hydrogen releases than currently required by the regulations particularly for smaller, low pressure containment designs.

This USI will investigate means to predict the quantity and release rate of hydrogen following degraded core accidents and various means to cope with large releases to the containment such as inerting of the containment or controlled burning. The potential effects of proposed hydrogen control measures on safety including the effects of hydrogen burns on safety related equipment will be investigated.

• RES INTERFACE INFORMATION
TO BE DEVELOPED

• ACRS INTERFACE INFORMATION
TO BE DEVELOPED

• TECHNICAL ASSISTANCE CONTRACTS
TO BE DEVELOPED

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED

• POTENTIAL PROBLEMS

• STATUS SUMMARY

A draft Task Action Plan (TAP) was issued for NRC comments on 4/02/82. Work has been suspended pending management decision on scope of TAP.

# INDICATES A CHANGE IN INFORMATION

PRESSURIZED THERMAL SHOCK (A-49)

AS OF WEEK ENDING AUGUST 20, 1982

KEY PERSONNEL

TASK MANAGER
ROY WOODS X24714

MPA ANALYST
M. ANDERSON X27201

TASK REVIEWERS

NAME	BRANCH
E. THROM	RSB/NRR
C. JOHNSON	RES
L. LOIS	CPB/DSI
J. STROSNIDER	RES
M. VAGINS	RES
R. JOHNSON	GIB/DST
R. KLECKER	MTEB/DE
N. RANDALL	MTEB/DE
G. VISSING	ORB-4/DL
J. CLIFFORD	PTRB/HFS

SCHEDULED COMPLETION

ORIGINAL Not Determined

CURRENT #Not Scheduled

• PROBLEM DESCRIPTION

This task was designated a USI by the NRC in December 1981.

Neutron irradiation of reactor pressure vessel weld and plate materials decreases the fracture toughness of the materials. The fracture toughness sensitivity to radiation induced change is increased by presence of certain materials such as copper. Decreased fracture toughness makes it more likely that, if a severe overcooling event occurs followed by or concurrent with high vessel pressure, and if a small crack is present on the vessels inner surface, that crack could grow to a size that might threaten vessel integrity.

Severe pressurized overcooling events are improbable since they require multiple failures and improper operator performance. However, certain precursor events have happened that could have potentially threatened vessel integrity if additional failures had occurred and/or if the vessel had been more highly irradiated. Therefore, the possibility of vessel failure due to a severe pressurized overcooling event cannot be ruled out.

• RES INTERFACE INFORMATION

A major portion of the work is being performed under a contract with Oak Ridge National Laboratory through the Division of Risk Analysis, RES (FIN # B0468).

Other major contributors are:

Primary System Integrity Research Program through the Division of Engineering Technology, RES and

Code Applications Program through the Division of Accident Evaluation, RES

• ACRS INTERFACE INFORMATION

Meetings have been held and will be scheduled as necessary with the Subcommittee on Metallic #Components and with the full ACRS. The next scheduled meetings are September 8 and 9, 1982, respectively.

• TECHNICAL ASSISTANCE CONTRACTS

Contract (B-2510) issued to PNL. PNL will provide their recommendations regarding the necessity of corrective actions, and will provide peer review of recommendations being considered by the NRC staff.

The following RES contracts are providing technical assistance to the PTS program. These are in addition to the technical assistance contracts which were initiated to specifically address the PTS issue and listed in the table below.

FIN No.	Description
B0119	HSST
B8133	LWR Pressure Boundary Integrity
B5988	Surveillance Dosimetry
B0415	Pressure Vessel Simulation
B6224	Dosimetry Meas. Data Base
B7026	JR Curve
A3215	Code Assessment and Application
A6047	Code Assessment and Applications
A7027	Analytical Res. in LWR Safety
A7217	TRAC Calc Assistance
B0468	Pressurized Thermal Shock

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
#B-2510	PNL	$400K	$375K-est.
A-7272	LANL	$210K-FY82	$ 80K-est.
		$210K-FY83	
A-3387	BNL	$210K	$ 94K-est.

• POTENTIAL PROBLEMS

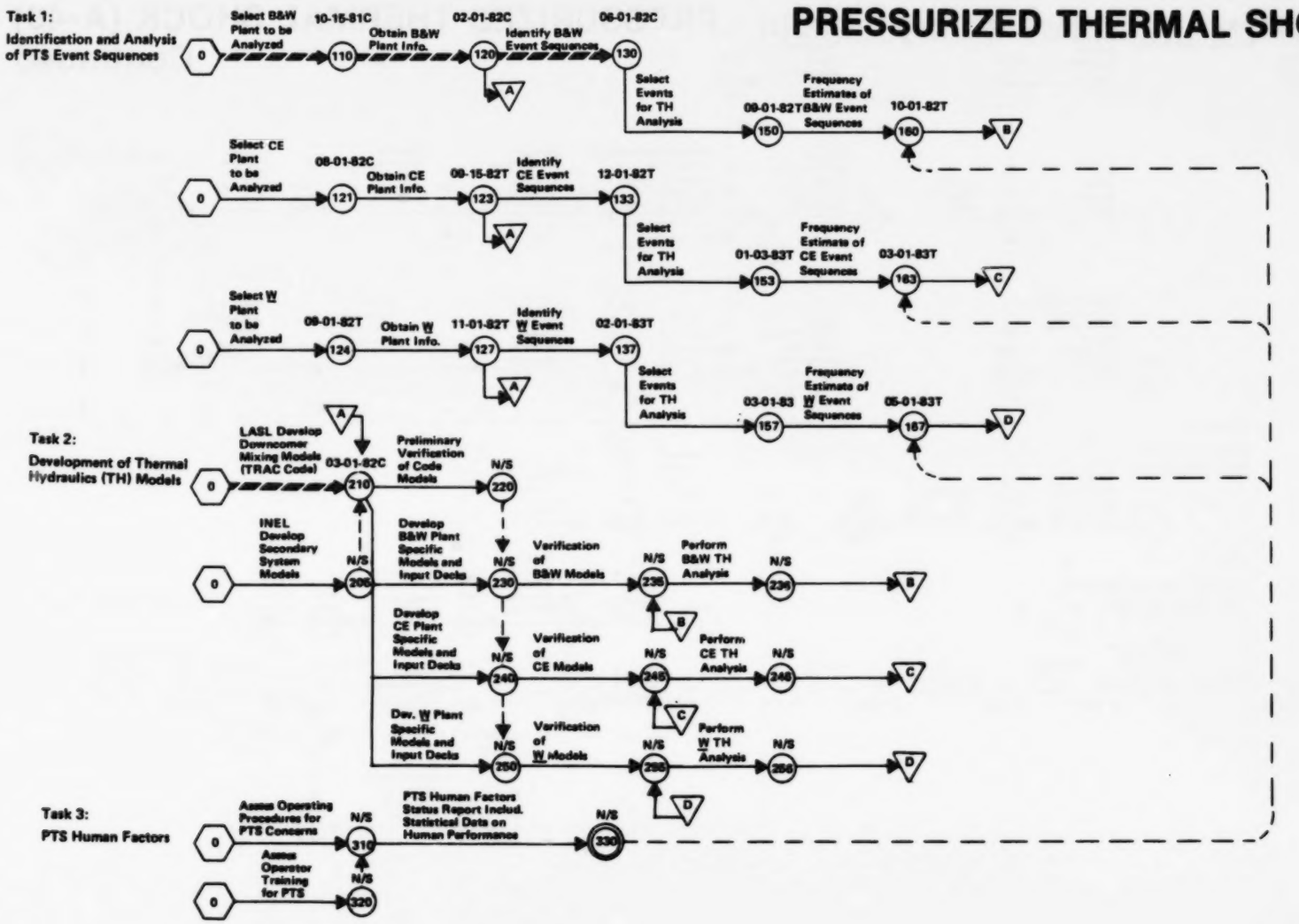
None.

• STATUS SUMMARY

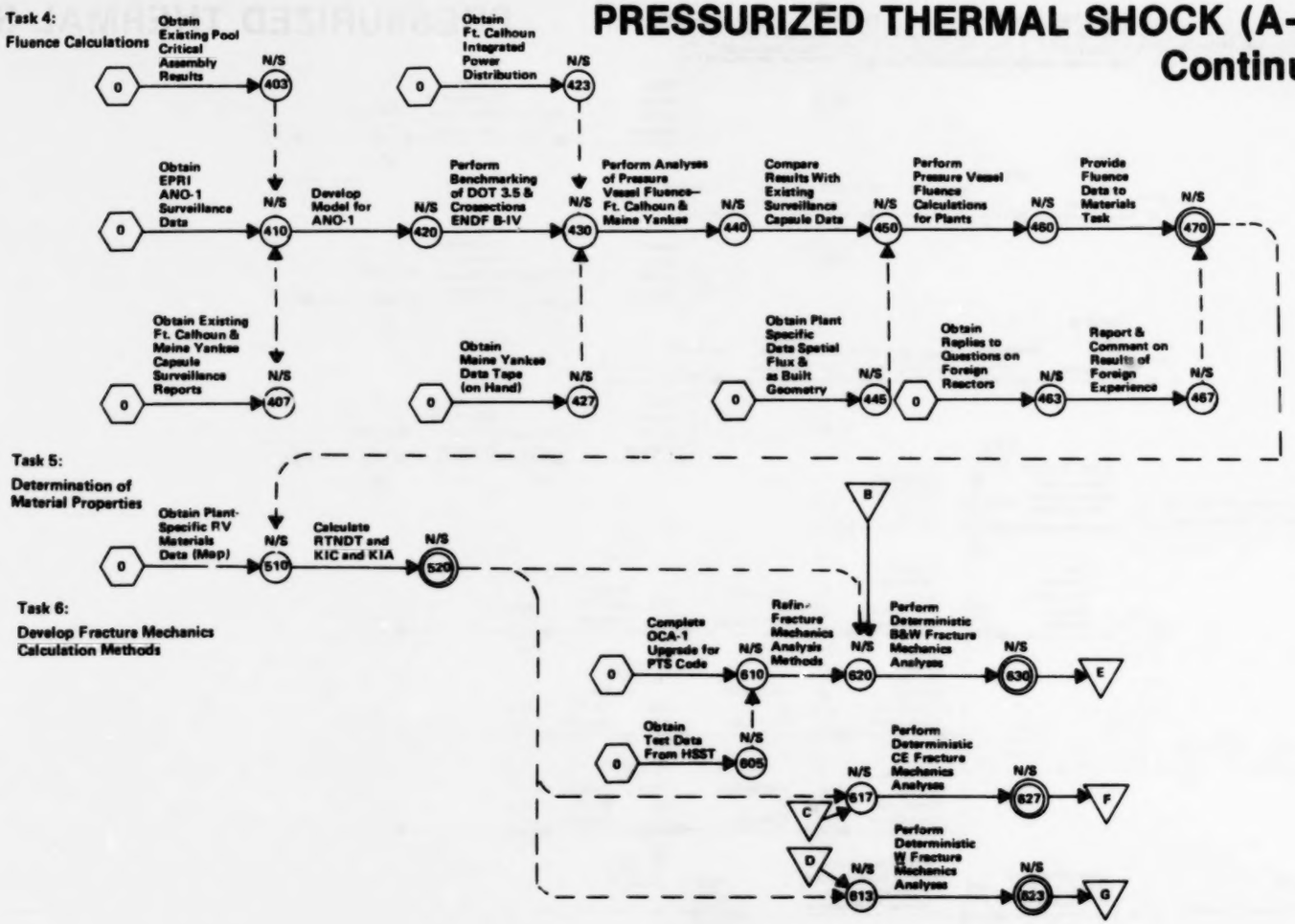
#Draft short-term position is under preparation and will be completed in Sept. 1982. Final PNL proposed position has been published as NUREG/CR-2837. ORNL has proposed the first three event sequences to be analyzed and they have been accepted by the staff. Review of the "150-day responses" has been completed. Results of the review are being integrated into the Sept. staff position report.

The final Task Action Plan for A-49 was approved and issued on March 26, 1982.

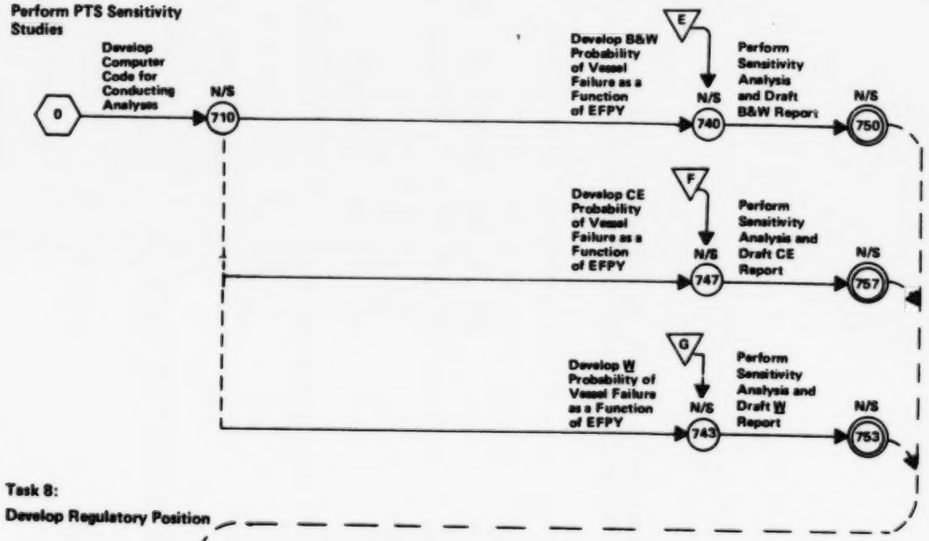
PRESSURIZED THERMAL SHOCK (A-49)



PRESSURIZED THERMAL SHOCK (A-49) Continued



PRESSURIZED THERMAL SHOCK (A-49)
Continued



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May 21, 1982 - August 20, 1982

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16 ABSTRACT (200 words or less)

Provide an overview of the status of the progress and plans for resolution of the generic tasks addressing "Unresolved Safety Issues" as reported to Congress.

17 KEY WORDS AND DOCUMENT ANALYSIS 17a DESCRIPTORS

17b IDENTIFIERS OPEN-ENDED TERMS

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UNRESOLVED SAFETY ISSUES SUMMARY

AQUA BOOK

Manuscript Completed: November 1982
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Prepared for: OFFICE OF RESOURCE MANAGEMENT

Prepared by: OFFICE OF RESOURCE MANAGEMENT
U.S. NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

FOREWORD

THE "UNRESOLVED SAFETY ISSUES" SUMMARY IS DESIGNED TO PROVIDE THE MANAGEMENT OF THE NUCLEAR REGULATORY COMMISSION WITH A QUARTERLY OVERVIEW OF THE PROGRESS AND PLANS FOR COMPLETION OF GENERIC TASKS ADDRESSING "UNRESOLVED SAFETY ISSUES" REPORTED TO CONGRESS PURSUANT TO SECTION 210 OF THE ENERGY REORGANIZATION ACT OF 1974 AS AMENDED. THIS SUMMARY UTILIZES DATA COLLECTED FROM THE OFFICE OF NUCLEAR REACTOR REGULATION, OFFICE OF NUCLEAR REGULATORY RESEARCH, AND THE NATIONAL LABORATORIES AND IS PREPARED BY THE OFFICE OF RESOURCE MANAGEMENT.

THE DEFINITION OF WHAT CONSTITUTES COMPLETION OF AN UNRESOLVED SAFETY ISSUE (USI) HAS RECENTLY BEEN EXPANDED TO INCLUDE THE IMPLEMENTATION OF THE TECHNICAL RESOLUTION. THIS IS IN ACKNOWLEDGEMENT OF THE FACT THAT REAL SAFETY BENEFITS OCCUR ONLY AFTER THE IMPLEMENTATION HAS TAKEN PLACE. IMPORTANT ELEMENTS OF THIS IMPLEMENTATION PHASE ARE:

1. THE PROVISION OF A PUBLIC COMMENT PERIOD FOLLOWING THE ISSUANCE OF A DRAFT NUREG REPORT INCORPORATING THE STAFF'S TECHNICAL RESOLUTION FOLLOWED BY A DISCUSSION AND DISPOSITION OF THE COMMENTS RECEIVED IN A FINAL NUREG REPORT.

2. THE PROVISION FOR INCORPORATION OF THE TECHNICAL RESOLUTION INTO THE NRC'S REGULATIONS, STANDARD REVIEW PLAN, REGULATORY GUIDES, OR OTHER NRC OFFICIAL GUIDANCE OR REQUIREMENTS AS APPROPRIATE.

3. THE PROVISION FOR APPLICATION OF THE TECHNICAL RESOLUTION TO INDIVIDUAL OPERATING PLANTS IN THE FORM OF HARDWARE OR DESIGN CHANGES, TECHNICAL SPECIFICATION CHANGE, AND/OR CHANGE TO OPERATING PROCEDURES AS APPROPRIATE.

THE SCHEDULES IN THIS BOOK INCLUDE A MILESTONE AT THE END OF EACH ACTION PLAN WHICH REPRESENTS THE INITIATION OF THE IMPLEMENTATION PROCESS BOTH WITH RESPECT TO INCORPORATION OF THE TECHNICAL RESOLUTION IN THE NRC OFFICIAL GUIDANCE OR REQUIREMENTS AND ALSO THE APPLICATION OF CHANGES TO INDIVIDUAL OPERATING PLANTS. THE SCHEDULE FOR IMPLEMENTATION WILL NOT NORMALLY BE INCLUDED IN THE TASK ACTION PLAN(S) FOR THE RESOLUTION OF A USI SINCE THE NATURE AND EXTENT OF THE ACTIVITIES NECESSARY TO ACCOMPLISH THE IMPLEMENTATION CANNOT NORMALLY BE REASONABLY DETERMINED PRIOR TO THE DETERMINATION OF A TECHNICAL RESOLUTION. THE PROGRESS AND STATUS FOR IMPLEMENTATION OF UNRESOLVED SAFETY ISSUES FOR WHICH A TECHNICAL RESOLUTION HAS BEEN COMPLETED ARE REPORTED SPECIFICALLY IN A SEPARATE TABLE PROVIDED IN THIS SUMMARY.

KARL KNIEL, CHIEF OF THE GENERIC ISSUES BRANCH, DIVISION OF SAFETY TECHNOLOGY/NRR, IS RESPONSIBLE FOR MANAGING THE GENERIC TASKS INCLUDED IN THIS SUMMARY.

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ABBREVIATIONS

AAB:	Accident Analysis Branch (Former NRR Branch)
AB:	Administration Branch, Training and Administration Staff (IE)
AC:	Alternating Current
ACRS:	Advisory Committee on Reactor Safeguards
AD:	Assistant Director
ADB:	Analysis and Development Branch, Division of Reactor Safety Research (RES)
AEB:	Accident Evaluation Branch, Division of Systems Integration (NRR)
AEOD:	Office of the Analysis and Evaluation of Operational Data
AIF:	Atomic Industrial Forum
APTS:	Action Plan Tracking System
ARL:	Alden Research Laboratory
ASB:	Auxiliary Systems Branch, Division of Systems Integration (NRR)
ASME:	American Society of Mechanical Engineers
ASTM:	American Society of Testing Materials
ATWS:	Anticipated Transient Without Scram
B&W:	Babcock and Wilcox Company
BNC:	Brookhaven National Conservatory
BNL:	Brookhaven National Laboratory
BOP:	Balance of Plant
BWR:	Boiling Water Reactor
CE:	Combustion Engineering, Incorporated
CEB:	Chemical Engineering Branch, Division of Engineering (NRR)
CFR:	Code of Federal Regulations
CO:	Condensation Isolation
CP:	Construction Permit
CPB:	Core Performance Branch, Division of Systems Integration (NRR)
CR:	Contractor Report
CRGR:	Committee for Review of Generic Requirements
CSB:	Containment Systems Branch, Division of Systems Integration (NRR)
DC:	Direct Current
DE:	Division of Engineering (NRR)
DFO:	Division of Facility Operations (RES)
DHFS:	Division of Human Factors Safety (NRR)
DHRS:	Decay Heat Removal System
DL:	Division of Licensing (NRR)
DOE:	U. S. Department of Energy
DOR:	Division of Operating Reactors (Former NRR Division)
DRA:	Division of Risk Analysis (RES)
DSI:	Division of Systems Integration (NRR)
DSS:	Division of Systems Safety (Former NRR Division)
DST:	Division of Safety Technology (NRR)
E:	Engineering
EB:	Enforcement Branch, Enforcement and Investigations Staff (IE)
ECC:	Emergency Core Cooling
EEB:	Environmental Engineering Branch, Division of Engineering (NRR)
EFPY:	Effective Full-Power Years
EG&G:	Edgerton, Germeshausen & Grier
EP:	Emergency Preparedness
EPRI:	Electric Power Research Institute
EQB:	Equipment Qualification Branch, Division of Engineering (NRR)
FIN:	Financial
FIRL:	Franklin Institute Research Laboratories
FSI:	Fluid Structure Interaction
FSTF:	Full-Scale Test Facility
FW:	Feedwater
GB:	Geosciences Branch, Division of Engineering (NRR)
GE:	General Electric
GIB:	Generic Issues Branch, Division of Safety Technology (NRR)
HFEB:	Human Factors Engineering Branch, Division of Health, Siting and Waste Mgmt. (RES)
HFS:	Division of Human Factors Safety (NRR)
HSST:	Heavy Section Steel Technology
ICBR:	Instrumentation and Control Branch, Division of Facility Operations (RES)
ICSB:	Instrumentation and Control Systems Branch, Division of Systems Integration (NRR)
IE:	Office of Inspection and Enforcement
IEEE:	Institute of Electrical and Electronics Engineers
INEL:	Idaho Nuclear Engineering Laboratory
IREP:	Integrated Reliability Evaluation Program
ISI:	In-Service Inspection
LER:	Licensee Event Report
LGB:	Licensing Guidance Branch, Division of Safety Technology (NRR)
LLL:	Lawrence Livermore Laboratory
LOCA:	Loss-of-Coolant Accident
LPP:	Lead Plant Program
LTP:	Long Term Program
LWR:	Light-Water Reactor
MARK I-III:	Containment Types for Boiling Water Reactors
MEB:	Mechanical Engineering Branch, Division of Engineering (NRR)
MEBR:	Materials Engineering Branch, Division of Engineering Technology (RES)
MIT:	Massachusetts Institute of Technology
MMRB:	Metallurgy and Materials Branch, Division of Reactor Safety Research (RES)
MSLB:	Main Steam Line Break
MTEB:	Materials Engineering Branch, Division of Engineering (NRR)
NDE:	Non-Destructive Examination
NREP:	Neutron Resonance Escape Probability
NRL:	Naval Research Laboratory
NRR:	Office of Nuclear Reactor Regulation
NSS:	Nuclear Steam System
NUREG:	Nuclear Regulation

ABBREVIATIONS

OG:	Owner's Group
OL:	Operating License
ORAB:	Operating Reactors Assessment Branch, Division of Licensing (NRR)
ORB:	Operating Reactors Branch, Division of Licensing (NRR)
ORNL:	Oak Ridge National Laboratory
OSD:	Office of Standards Development (Former NRC Office)
OTSG:	Once-Through Steam Generator
PASNY:	Power Authority of the State of New York
PNL:	Pacific Northwest Laboratory (Battelle)
PRA:	Probabilistic Risk Assessment
PS:	Plant Systems
PSB:	Power Systems Branch, Division of Systems Integration (NRR)
PSU:	Plant Systems Unit (AEOD)
PSYB:	Plant Systems Branch (Former NRR Branch)
PTRB:	Procedures and Test Review Branch, Division of Human Factors Safety (NRR)
PWR:	Pressurized Water Reactor
Q:	Quality Control or Assurances
RAB:	Radiological Assessment Branch, Division of Systems Integration (NRR)
RCIC:	Reactor Core Isolation Cooling
REF:	Reference
RES:	Office of Nuclear Regulatory Research
RFP:	Request for Proposal
RHR:	Residual Heat Removal
RPV:	Reactor Pressure Vessel
PRA:	Probability and Risk Assessment
RRAB:	Reliability and Risk Assessments Branch (NRR)
RRRC:	Regulatory Requirements Review Committee
RS:	Reactor Safety (Former NRR Branch)
RSB:	Reactor Systems Branch, Division of Systems Integration (NRR)
RSR:	Reactor Safety Research
RSSMAP:	Reactor Safety Study Methodology Application Program
SAI:	Science Applications, Inc.
SCC:	Stress-Corrosion Cracking
SEB:	Structural Engineering Branch, Division of Engineering (NRR)
SEP:	Systematic Evaluation Program
SEPB:	Systematic Evaluation Program Branch, Division of Licensing (NRR)
SER:	Safety Evaluation Report
SG:	Steam Generator
SGWH:	Steam Generator Water Hammer
SNL:	Sandia National Laboratory
SRP:	Standard Review Plan
SRV:	Safety Relief Valve
SSE:	Safe Shutdown Earthquake
SSPB:	Standards Special Projects Branch (NRR)
STP:	Short-term Program
TAP:	Task Action Plan
TH:	Hot Leg Temperature
TM:	Task Manager
TMI:	Three Mile Island
USI:	Unresolved Safety Issue
W:	Westinghouse Electric Corporation
WH:	Water Hammer
WUTA:	Washington University Technology Associates

PROGRAM OVERVIEW

PROJECTED DATES FOR ISSUING NRC STAFF REPORTS

Key:
(1) LPP - Lead Plant Program
(2) LTP - Long Term Program
N/S - Not Scheduled

USI NO.	TITLE	AUG. 20, 1982 AQUA BOOK	CURRENT	REMARKS
A-1,	Water Hammer	August 1, 1983	November 30, 1983	NUREG/CR-2781 was issued in July 1982. Development of a technical findings NUREG is underway. This draft NUREG is being used to develop recommendations to CRGR for resolution.
A-3, A-4, A-5,	Steam Generator Tube Integrity	March 30, 1983	March 30, 1983	The draft NUREG report has been reviewed by the technical branches and their comments incorporated. The revised report is being prepared for management review and then will be issued for public comment. The NRC has formed a Task Force to prepare proposed requirements.
A-11,	Reactor Vessel Materials Toughness	October 15, 1982	October 15, 1982	NUREG-0774, Rev. 1, was completed and submitted for review. This revision addresses public comments received on the For Comment report. This report was issued in October 1982, thereby resolving this Unresolved Safety Issue.

PROGRAM OVERVIEW

PROJECTED DATES FOR ISSUING NRC STAFF REPORTS

USI NO.	TITLE	AUG. 20, 1982 AQUA BOOK	CURRENT	REMARKS
A-12,	Steam Generator & Reactor Coolant Pump Supports	December 30, 1982	June 6, 1983	Meetings were held with EPRI and Industry groups in December 1980, to resolve use of fracture mechanics methods for qualification of supports. NRC reviewer's comments on the final NUREG-0577 were not received until mid-January 1982, causing a delay in issuing the final report. Comments made by EPRI and the AIF subcommittee on material requirements have been incorporated into the final NUREG.
A-17,	Systems Interactions	See Item II.C.3 in Action Plan Tracking System	See Item II.C.3 in Action Plan Tracking System	Work planned under TAP A-17 will be performed under TMI Action Plan item II.C.3, Systems Interaction. The status of this item is reported in the quarterly updates of the TMI Action Plan Tracking System (APTS).
A-39,	SRV Pool Dynamic Loads	Mark II-September 30, 1982 Mark III-October 15, 1982	Mark II-September 30, 1982 Mark III-October 14, 1982	NUREG-0763 on in-plant SRV tests was completed and issued in June 1981. NUREG-0783 on Pool Temperature Limits was issued in October 1981. NUREG-0802 on SRV Quencher Loads was issued in October 1982. These three reports serve as the technical resolution of USI A-39.
A-40,	Seismic Design Criteria	July 15, 1983	March 16, 1984	A draft standard review plan change has been prepared and is under management review.
A-43,	Containment Emergency Sump	July 15, 1983	July 15, 1983	All technical support (NUREG/CRs) reports have been issued except the ARL report on insulation debris transport and screen blockage effects. A resolution NUREG is in the final stages of preparation and will undergo management concurrence. A CRGR briefing will be held November 24, 1982 to discuss the resolution of this USI.

PROGRAM OVERVIEW
PROJECTED DATES FOR ISSUING NRC STAFF REPORTS

USI NO.	TITLE	AUG. 20, 1982 AQUA BOOK	CURRENT	REMARKS
A-44,	Station Blackout	October 15, 1983	December 15, 1983	Draft contractor technical reports were provided in May 1982. Final reports are now in preparation.
A-45,	Shutdown Decay Heat Removal Requirements	October 30, 1985	October 30, 1985	A Task Action Plan was issued on October 7, 1981, and revised on June 2, 1982. A technical assistance contract package was approved on April 9, 1982 and implemented with Sandia on May 10, 1982.
A-46,	Seismic Qualification of Equipment in Operating Plants	May 1, 1984	April 30, 1984	This program was redirected to be made compatibile with the Equipment Qualification Program Plan. The final Task Action Plan has been issued. Work has been initiated on all tasks with all contractors well into their subtasks.
A-47,	Safety Implications of Control Systems	N/S	N/S	An approved Task Action Plan was issued on Sept. 27, 1982. Technical Assistance contracts have been established with ORNL and INEL.
A-48,	Hydrogen Control Measures and Hydrogen Burns on Safety Equipment	N/S	N/S	A draft Task Action Plan was issued on April 2, 1982 for NRC comments. Work has been suspended pending management decision on the scope of the TAP.
A-49,	Pressurized Thermal Shock	N/S	N/S	A-49 was designated by the Commission as an USI on December 28, 1981. The Task Action Plan was approved and issued on March 26, 1982. The draft short-term position is under preparation and will be completed in November 1982. As a result of the findings of the short-term program, the A-49 network is going to be extensively revised.

USI'S FOR WHICH TECHNICAL RESOLUTION IS COMPLETE

USI NO.	TITLE	DATE COMPLETED	REPORTS PUBLISHED	IMPLEMENTATION STATUS	IMPLEMENTATION TASK MANAGER/ TELEPHONE
A-2,	ASYMMETRIC BLOWDOWN LOADS ON REACTOR PRIMARY COOLANT SYSTEMS	DEC. 1980	NUREG-0609	ALL PWR PLANT ASSESSMENTS FOR ASYMMETRIC LOADS HAVE BEEN RECEIVED AND ARE CURRENTLY BEING EVALUATED BY STAFF AND CONTRACTOR EG&G. AS A BASIS FOR THIS EVALUATION, STAFF IS IMPLEMENTING CRITERIA DEVELOPED AS PART OF THE TASK ACTION PLAN A-2. THESE CRITERIA ARE INCLUDED IN NUREG-0609. B&W AND CE OWNERS' GROUPS HAVE RESPONDED TO THE QUESTIONS RAISED AS A RESULT OF OUR INITIAL REVIEW OF THEIR SUBMITTALS ON THE ASSESSMENT OF THE PRIMARY SYSTEM COMPONENTS AND SUPPORTS TO ASYMMETRIC LOCA LOADS. THESE RESPONSES ARE CURRENTLY BEING REVIEWED AND THE REVIEW IS EXPECTED TO BE COMPLETED BY NOVEMBER 1983. THE WESTINGHOUSE OWNERS GROUPS EXPECT A RESOLUTION OF THIS ISSUE ON THE BASIS OF 'LEAK BEFORE BREAK' CRITERIA. AN INTERNAL REPORT, RECOMMENDING ACCEPTANCE OF THE 'LEAK BEFORE BREAK' CONCEPT ISSUED BY THE STAFF IN FEBRUARY 1981, HAS BEEN DISCUSSED WITH THE ACRS AND IS UNDER REVIEW BY NRR MANAGEMENT PRIOR TO REVIEW BY CRGR.	JIM SHEA 492-7231
A-6,	MARK I SHORT TERM PROGRAM	DEC. 1977	NUREG-0408	COMPLETE - ALL PLANT UNIQUE ANALYSES AND EQUIPMENT MODIFICATIONS AS REQUIRED WERE REVIEWED AND ACCEPTED AND APPROPRIATE TECHNICAL SPECIFICATION CHANGES WERE MADE.	- - -

USI'S FOR WHICH TECHNICAL RESOLUTION IS COMPLETE

USI NO.	TITLE	DATE COMPLETED	REPORTS PUBLISHED	IMPLEMENTATION STATUS	IMPLEMENTATION TASK MANAGER/ TELEPHONE
A-7,	MARK I LONG TERM PROGRAM	JULY 1980 AUGUST 1982	NUREG-0661 NUREG-0661, SUPPL. No. 1	LICENSEES ARE IN THE PROCESS OF DESIGNING AND INSTALLING MODIFICATIONS TO MEET THE COMMISSION'S ORDER DATE FOR EACH OPERATING PLANT. THE LICENSEES ARE IN THE PROCESS OF PREPARING PLANT-UNIQUE ANALYSIS FOR SUBMITTAL TO THE STAFF FOR POST IMPLEMENTATION AUDIT REVIEW FOR COMPLIANCE WITH THE ACCEPTANCE CRITERIA CONTAINED IN APPENDIX A TO NUREG-0661. THREE LICENSEES/APPLICANTS HAVE SUBMITTED THEIR PLANT-UNIQUE ANALYSIS.	BYRON SIEGEL 492-7534
A-8,	MARK II CONTAINMENT POOL DYNAMIC LOADS	AUGUST 1981	NUREG-0808	THE REQUIREMENTS RECOMMENDED IN NUREG-0808 ARE BEING IMPLEMENTED DURING THE OPERATING LICENSE REVIEW FOR EACH PLANT WITH A MARK II CONTAINMENT. THESE REQUIREMENTS WILL ALSO BE INCLUDED AS AN ADDITION TO THE APPROPRIATE SECTION OF THE STANDARD REVIEW PLAN.	
A-9,	ATWS	SEPT. 1980	NUREG 0460, VOL. 4 46FR57521	TWO VERSIONS OF THE PROPOSED ATWS RULE HAVE BEEN ISSUED FOR COMMENT; THE STAFF VERSION AND AN ALTERNATE VERSION OF THE RULE PREPARED AT THE DIRECTION OF COMMISSIONER HENDRIE. FOLLOWING THE COMMENT PERIOD AND CONSIDERATION OF PUBLIC COMMENTS, A FINAL RULE WILL BE PREPARED FOR COMMISSION CONSIDERATION. A TASK FORCE HAS BEEN ASSIGNED TO FORM A BASIS FOR THE FINAL RULE.	DAVID PYATT, RES 443-5960

USI'S FOR WHICH TECHNICAL RESOLUTION IS COMPLETE

USI NO.	TITLE	DATE COMPLETED	REPORTS PUBLISHED	IMPLEMENTATION STATUS	IMPLEMENTATION TASK MANAGER/ TELEPHONE
A-10,	BWR FEEDWATER NOZZLE CRACKING	NOV. 1980	NUREG-0619	RESPONSES FROM LICENSEES TO AN IMPLEMENTATION LETTER HAVE BEEN RECEIVED AND RECOMMENDED TREATMENT OF THESE RESPONSES HAVE BEEN SUBMITTED TO NRC MANAGEMENT. ADDITIONAL INFORMATION HAS BEEN REQUESTED OF LICENSEES. IMPLEMENTATION ACTION IS PROCEEDING. THIRTEEN PLANTS HAVE RECEIVED LETTERS ACCEPTING THEIR PROPOSED MODIFICATION PLANS. TEN PLANTS ARE BEING REVIEWED AND NEGOTIATIONS ARE UNDER WAY TO OBTAIN LICENSEE COMMITMENTS TO IMPLEMENT ALL NUREG-0619 GUIDANCE. LACROSSE, BIG ROCK POINT, AND DRESDEN 1 DO NOT HAVE SUSCEPTIBLE PLANT SYSTEM CONFIGURATIONS AND ARE CONSIDERED COMPLETE WITH REGARD TO THIS ACTION. HUMBOLDT BAY, BY VIRTUE OF ITS STATUS (SHUTDOWN, NO FORSEEABLE RESTART), IS ALSO CONSIDERED COMPLETE.	PAUL O'CONNOR 492-7215
A-11,	REACTOR VESSEL MATERIALS TOUGHNESS	OCTOBER 15, 1982	NUREG-0744, VOLS. I AND II	GENERIC LETTER TRANSMITTING THIS NUREG REPORT IS BEING PREPARED BY DL.	AL DEAGAZIO 492-8945
A-24,	QUALIFICATION OF CLASS 1E SAFETY RELATED EQUIPMENT	AUG. 1981	NUREG-0588	EQB HAS THE LEAD IN IMPLEMENTING THE POSITIONS IDENTIFIED IN THE REPORT.	JOSE' CALVO 492-8563
A-26,	REACTOR VESSEL PRESSURE TRANSIENT PROTECTION	SEPT. 1978	NUREG-0224	ALL PLANTS WERE REQUESTED TO PROVIDE AN OVERPRESSURE PREVENTION SYSTEM THAT WOULD BE USED WHENEVER THE PLANT WAS IN A COLD SHUTDOWN CONDITION. ALL PWRS IMPLEMENTED THEIR SYSTEMS WITH PRELIMINARY APPROVAL FROM THE NRC, AND A COMPLETE REVIEW TOOK PLACE ON A POST-IMPLEMENTATION BASIS. THE REVIEW CONSISTS OF THREE PARTS: A) ELECTRICAL ASPECTS: B) REACTOR SYSTEM ASPECTS; C) TECHNICAL SPECIFICATIONS. PLANTS REMAINING TO BE EVALUATED ARE BEING REVIEWED BY CONTRACT WITH EG&G. THIS INCLUDES FIVE OF THE TWELVE REMAINING PLANTS. FOR THE OTHER SEVEN PLANTS, SIX SER'S HAVE BEEN COMPLETED AND ARE IN THE PROCESS OF BEING ISSUED.	AL DEAGAZIO 492-8945

USI'S FOR WHICH TECHNICAL RESOLUTION IS COMPLETE

USI NO.	TITLE	DATE COMPLETED	REPORTS PUBLISHED	IMPLEMENTATION STATUS	IMPLEMENTATION TASK MANAGER/ TELEPHONE
A-31,	RESIDUAL HEAT REMOVAL REQUIREMENTS	1978	NO FORMAL REPORT SRP 5.4.7 REVISION	RRRC APPROVED IMPLEMENTATION PLAN JANUARY 31, 1978. IMPLEMENTATION FOR OPERATING REACTORS NOT COMPLETE.	TOM NOVAK 492-7817
A-36,	CONTROL OF HEAVY LOADS NEAR SPENT FUEL	JULY 1980	NUREG-0612	IMPLEMENTATION REQUIREMENTS WERE ISSUED TO ALL LICENSEES BY LETTER DATED DECEMBER 22, 1980. THE LETTER REQUESTED; INTERIM ACTIONS TO BE COMPLETED IN 90 DAYS, A PHASE I ACTION (REPORT, CONFIRMATION AND JUSTIFICATION) IN SIX MONTHS AND PHASE II (SPECIFIC REQUIREMENTS) IN NINE MONTHS. ALL LICENSEES (EXCEPT ONE) HAVE RESPONDED TO THE DECEMBER 22, 1980 GENERIC LETTER AND THEIR RESPONSES HAVE BEEN EVALUATED. NINE PLANTS HAD REQUESTED EXTENSIONS TO THE SEPTEMBER 22, 1981 RESPONSE DATE. KEWAUNEE IS SCHEDULED TO RESPOND BY 12/14/82.	BUD REQUA 492-7877
A-39,	DETERMINATION OF SAFETY RELIEF VALVE (SRV) POOL DYNAMIC LOADS AND TEMPERATURE LIMITS FOR BWR CONTAINMENT	MARK I-02-29-80 MARK II-09-30-82 MARK III-10-14-82	NUREG-0763 NUREG-0783 NUREG-0802	GENERIC LETTER TRANSMITTING ALL THREE NUREGS TO BWR APPLICANTS AND LICENSEES IS IN THE DL CONCURRENCE CHAIN.	BYRON SIEGEL 492-7534
A-42,	PIPE CRACKS IN BOILING WATER REACTORS	JULY 1980	NUREG-0313 REV. 1	IN FEBRUARY 1981, NUREG-0313, REV. 1 WAS ISSUED TO ALL HOLDERS OF BWR OPERATING LICENSES OR CONSTRUCTION PERMITS AND TO ALL APPLICANTS FOR OPERATING LICENSES. BY JULY 1, 1981, THE APPLICANTS/LICENSEES WERE TO PROVIDE THEIR PROGRAM FOR REPLACEMENT OF SERVICE SENSITITIVE LINES AND WELDS, THEIR PROGRAM FOR AUGMENTED INSERVICE INSPECTION, THEIR PROGRAM FOR IMPROVING THE WATER CHEMISTRY ENVIRONMENT AND INCOPORATION OF ADEQUATE LEAK DETECTION CAPABILITY. ALL LICENSEES HAVE RESPONDED AND THEIR REPLIES ARE BEING EVALUATED BY A TECHNICAL SUPPORT CONTRACTOR (INEL). INITIAL TER'S ARE SCHEDULED TO BE COMPLETED IN THE SECOND AND THIRD QUARTERS OF FY83.	DICK CLARK 492-7162

WATER HAMMER (A-1)

AS OF WEEK ENDING NOVEMBER 19, 1982

KEY PERSONNEL

TASK MANAGER
ALECK SERKIZ (DST) X24217

MPA ANALYST
A. LANIER X28102

TASK REVIEWERS

NAME	BRANCH
S. MCKAY	PTRB/DHFS
C. GRAVES	RSB/DSI

SCHEDULED COMPLETION

1978 ANNUAL REPORT 12-31-80

CURRENT # 11-30-83

• PROBLEM DESCRIPTION

Since 1969 there have been over 150 incidents involving water hammers in BWRs and PWRs reported. The water hammers (or steam hammers) have involved steam generator feedrings and piping, the RHR system, ECC systems, and containment spray, service water, feedwater and steam lines. The incidents have been attributed to such causes as rapid condensation of steam pockets, steam-driven slugs of water, pump startup with partially empty lines, and rapid valve motion. Most of the damage reported has been relatively minor, involving pipe hangers and restraints; however, there have been several incidents which have resulted in piping and valve damage.

Task Action Plan, TAP A-1, subdivides resolution of this Unresolved Safety Issue (USI) into the following tasks:

Task 1: Water Hammer Summary Reports

Task 2: Resolution of Water Hammer Issue

Task 3: Water Hammer Position for Operating Reactors

Task 4: Water Hammer Safety Studies

Task 1 and 4 have been completed and the reports are on the network charts which follow. The near-term position for operating reactors is to permit continued operation since, as noted above, damage attributed to water hammer has been relatively minor. The long-term aspect of Task 3, is an integral part of the resolution of this issue.

Task 2 represents the resolution aspect of this USI and will result in the final resolution.

• RES INTERFACE INFORMATION

None.

• ACRS INTERFACE INFORMATION

A presentation was made to the ECCS Subcommittee on 12/03/81, plans for resolution of A-1 were presented.

• TECHNICAL ASSISTANCE CONTRACTS

Resolution efforts for A-1 are being conducted under FIN No. A6451 at INEL. Both EG&G-Idaho and Quadrex staff are being utilized.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
FY 81:			
A6451	EG&G-Idaho	$120,000	$ 39,355
FY 82:			
A6451	EG&G-Idaho		
	10/81	$120,000	$ 44,069
	11/81	145,000	56,285
	12/81	305,000	67,613
	01/82	305,000	137,194
	02/82	305,000	178,590
	03/82	305,000	207,411
	04/82	305,000	232,563
	05/82	330,000	244,984
	06/82	330,000	253,151
#	07/82	330,000	240,000
#	08/82	330,000	249,000
#	09/82	330,000	263,000

• POTENTIAL PROBLEMS

• STATUS SUMMARY

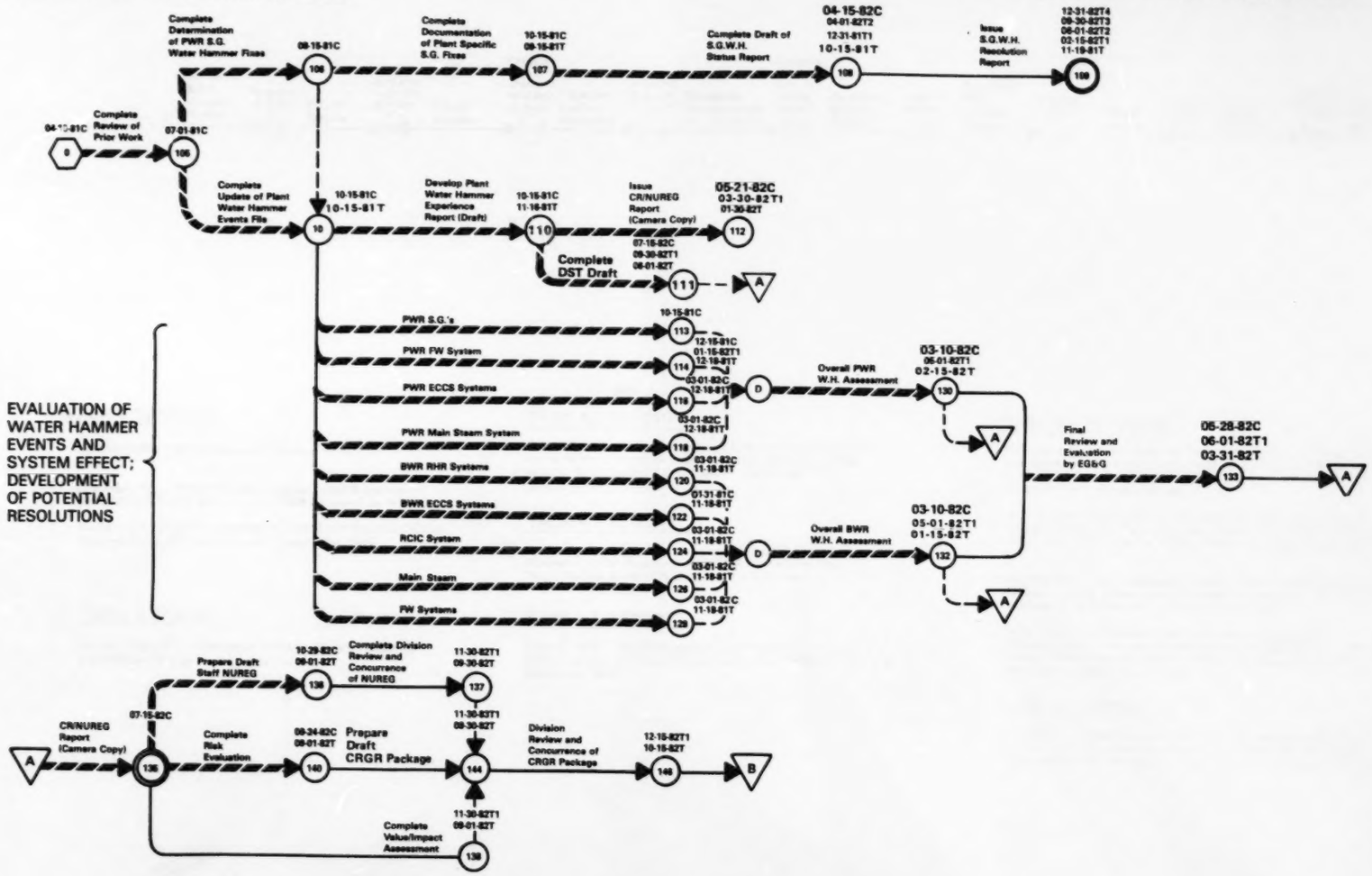
NUREG/CR-2781, "Evaluation of Water Hammer Events in Light Water Reactor Plants," July 1982 was issued, thus completing Milestone No. 136, Task 2.0.

Development of the staff's technical findings NUREG is nearing completion. This NUREG will be used to develop recommendations to the CRGR for resolution of USI A-1.

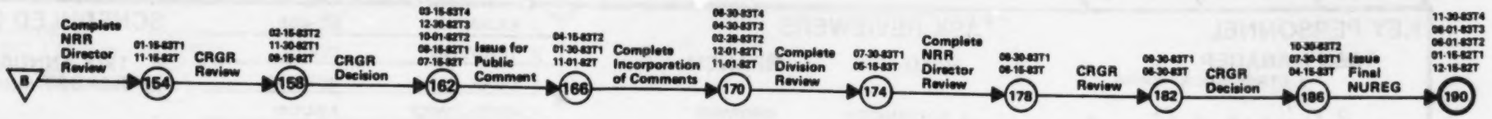
# INDICATES A CHANGE IN INFORMATION

TASK 2.0
RESOLUTION OF A-1, WATER HAMMER

WATER HAMMER (A-1)



WATER HAMMER (A-1) Continued



TASK 1 REPORTS

NUREG-0582, "Water Hammer in Nuclear Power Plants" July 1979

Chapman, R.L., "Water Hammer Studies," EG&G Interim Rpt., CAAP-TR-053 (Rev. 1), July 1980 (Prepared for NRC Use)

NUREG/CR-2059, "Compilation of Actual and Suspected Water Hammer Events in Nuclear Power Plants," May 1982.

TASK 2 REPORT

NUREG/CR-2781, "Evaluation of Water Hammer Events in Light Water Reactor Plants," July 1982.

TASK 4.1 REPORT

Chapman, R.L., et al., "Review and Evaluation of Actual and Potential Water Hammer Events in Nuclear Plants," EG&G Interim Rpt., CAAP-TR-042 (Rev. 1), September 1979 (Prepared for NRC Use)

TASK 4.2 REPORT

Watkins, J.C., Berry, R.A., "A State-of-the-Art Literature Review of Water Hammer," EG&G Interm Interim Rpt., RE-A-70-044, April 1979 (Prepared for NRC Use)

TASK 4.3 REPORT

Saha, P., et al., "An Evaluation of Condensation - Induced Water Hammer in Preheat Steam Generators, "NUREG/CR-1606, September 1980

TASK 4.4 REPORTS

Williamson, R.L., "An Analysis Tool for Predicting the Transient Hydrodynamics Resulting from the Rapid Filling of Voided Piping Systems," EG&G Interim Rpt. RE-E-79-009, February 1979 (Prepared for NRC Internal Use)

Morton, D.K., "An Analytical Procedure for Performing Structural Analyses of Nuclear Piping Systems Subjected to Fluid Transients," EG&G Interim Rpt., RE-E-79-013, February (Prepared for NRC Internal Use)

Dammie, P.N., "An Investigation of the Steam Void Collapse Water Hammer Initiating Mechanism," EG&G Interim Rpt., RE-A-78-229, February 1979 (Prepared for NRC Internal Use)

Berry, R.A., "An Analysis Tool for Predicting Transient Hydrodynamics in Nuclear Piping Systems Containing Swing Check Valves," EG&G Rpt. RE-A-78-261 (Rev. 2), September 1979 (Prepared for NRC Internal Use)

TASK 4.5 REPORT

Saffell, B.F., "Supplemental Water Hammer Analysis and System Review," EG&G preliminary Rpt. EGG-CAAP-5133, July 1980 (Prepared for NRC for Use as a Preliminary or Working Document)

WEST., CE & B&W STEAM GENERATOR TUBE INTEGRITY (A-3, A-4, A-5)

AS OF WEEK ENDING NOVEMBER 19, 1982

KEY PERSONNEL

TASK MANAGER
J. STROSNIDER X35903

P. Norian J. Strosnider

MPA ANALYST
B. ST. MARY X28330

TASK REVIEWERS

NAME	BRANCH
J. STROSNIDER	MEBR/RES
W. PASEDAG	AEB/DSI
C. PARSZEWSKI	CEB/DE
J. RAJAN	MEB/DE
B. TUROVLIN	CEB/DE
F. ODAR	ADB/RSR
F. AKSTULEWICZ	AEB/DSI
J. ZWOLINSKI	HFEB/DHFS
L. FRANK	MEBR

SCHEDULED COMPLETION

1978 ANNUAL REPORT	Early 1980
CURRENT	03-30-83

• PROBLEM DESCRIPTION

Pressurized water reactor steam generator tube integrity can be degraded by corrosion induced wastage, cracking, reduction in tube diameter (denting) and vibration induced fatigue cracks. The primary concern is the capability of degraded tubes to maintain their integrity during normal operation and under accident conditions (LOCA or a main steam line break) with adequate safety margins.

Westinghouse and Combustion Engineering steam generator tubes have suffered degradation due to wastage and stress corrosion cracking. Both types of degradation have been decreased by conversion from phosphate to an all-volatile secondary water treatment. Degradation due to denting which leads to primary side stress corrosion cracking continues to be a problem.

B&W's once-through steam generators (OTSG's) were relatively free of trouble prior to the first tube leak incident at Oconee Unit 3 in July, 1976. Since then, all three Oconee units have experienced tube leak incidents. The leaks at the Oconee units are the result of cracks of unknown origins propagated in the circumferential direction by flow induced vibration and have been limited to tubes located adjacent to the open tube inspection lane.

A second form of degradation characterized as an erosion-cavitation phenomena has been observed at Oconee and other B&W units.

• RES INTERFACE INFORMATION

A. RES has funded, at the request of NRR, a major confirmatory program at PNL. The activity of this program consists of tests to verify the burst and cyclic strengths of degraded steam generator tubes and the leak rate data.
B. RES is funding a program addressing the factors which determine Inconel 600 susceptability to stress corrosion cracking in primary water. Metal condition, chemistry, temperature, stress and environment will be considered.

• ACRS INTERFACE INFORMATION

ACRS Subcommittee meeting is scheduled for 01/28/83.

• TECHNICAL ASSISTANCE CONTRACTS

The following technical assistance contracts are generic in nature and will be applicable to the three Category "A" Technical Activities (A-3, A-4, and A-5) related to PWR steam generators.

A SANDIA — Provide statistical analysis of steam generator tube failures in operating reactors in order to establish the bases for the sampling plan for inservice inspection. Completed.
B BNL — Provide necessary computer code and perform parametric evaluation of effects of tube failures concurrent with MSLB. Completed.
C BNL — Provide technical consultation and assistance to review information in areas of water chemistory and corrosion analysis, stress and/or burst strength calculations. Completed.
D PNL — Provide cost/benefit evaluation of ISI plans. Completed.
E PNL — Evaluate environmental consequences of multiple tube failures concurrent with MSLB. Completed.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
2314	PNL	$75,000	$75,000
B2315	PNL	$95,000	$95,000

• POTENTIAL PROBLEMS

Management review and concurrence and CRGR approval are the only remaining actions necessary for # issuance of the staff report, "NRC Recommendations Concerning Steam Generator Tube Degradation and Rupture Events."

• STATUS SUMMARY

The NRC has formed a Task Force to prepare its proposed requirements regarding steam generator tube integrity. These requirements will include new concerns resulting from the Ginna tube failure (such as loose parts in the secondary system and plant response to SG tube failures) and also corrosion related failure mechanisms. The recommendations prepared by the staff under USI A-3, 4, 5 were primarily concerned with corrosion mechanisms such as wastage and denting. Consequently, as discussed with the Commission on June 30, 1982, the requirements from the USI program will be incorporated in the overall set of requirements being developed to address tube failures.

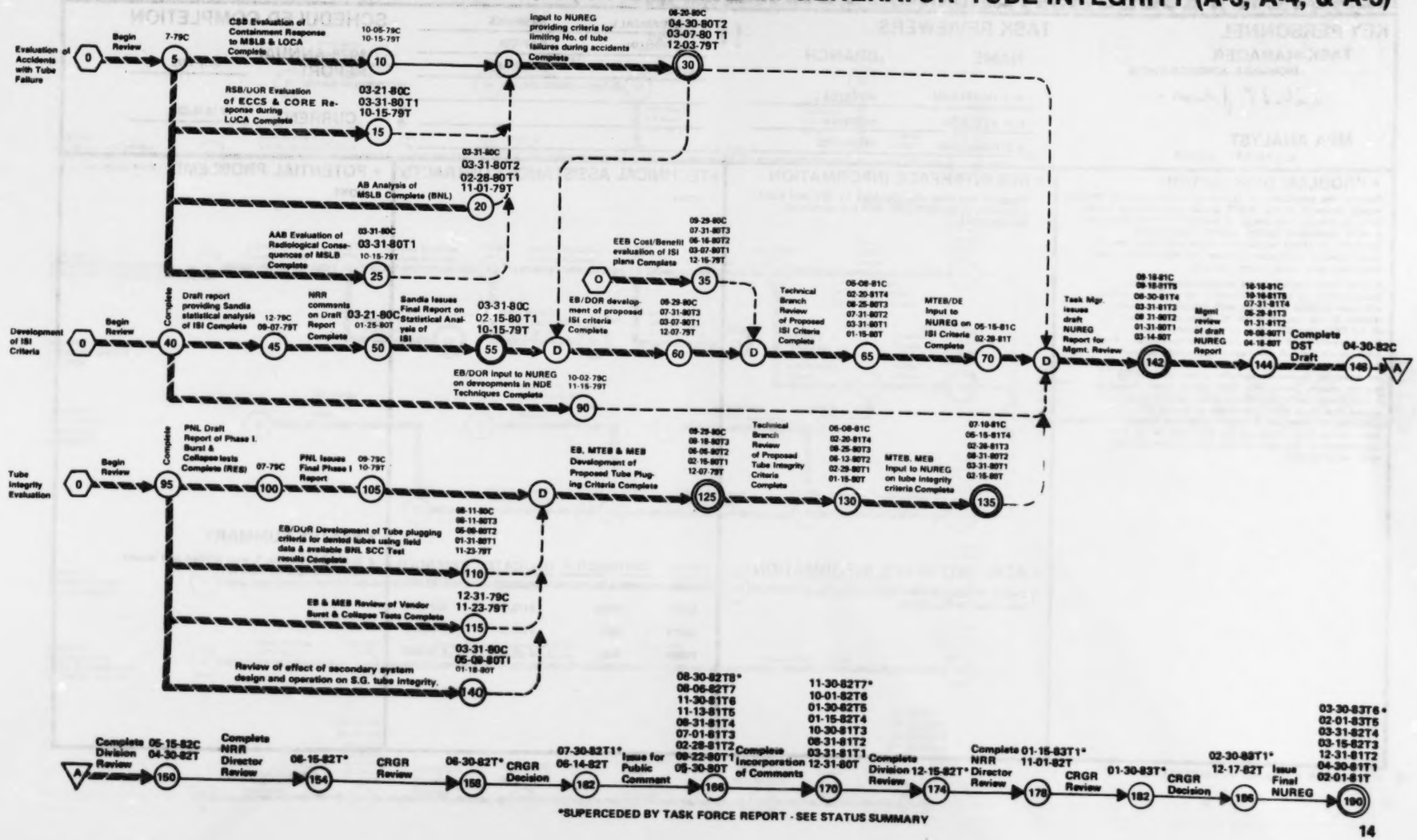
The proposed requirements were discussed with the Steam Generator Owners' Group on July 29, 1982; comments from the group were received September 30, 1982.

Science Applications, Inc. performed a Value/Impact analysis of the recommendations and issued their final draft report on September 23, 1982.

The staff's revised recommendations are in preparation for management review.

# INDICATES A CHANGE IN INFORMATION

WEST., CE & B&W STEAM GENERATOR TUBE INTEGRITY (A-3, A-4, & A-5)



*SUPERCEDED BY TASK FORCE REPORT - SEE STATUS SUMMARY

REACTOR VESSEL MATERIALS TOUGHNESS (A-11)

AS OF WEEK ENDING NOVEMBER 19, 1982

KEY PERSONNEL

TASK MANAGER
RICHARD E. JOHNSON X24715

Richard E. Johnson

MPA ANALYST
L. LANIER X28012

TASK REVIEWERS

NAME	BRANCH
W.S. HAZELTON	MTEB/DE
R.W. KLECKER	MTEB/DE
J. STROSNIDER	MEBR/RES
P.N. RANDALL	MEBR/RES
C.S. HINSON	RAB/DSI
C. SERPAN	MEBR/RES

SCHEDULED COMPLETION

1978 ANNUAL REPORT	July 1979
CURRENT	# 10-15-82C

• PROBLEM DESCRIPTION

Because the possibility of failure of nuclear reactor pressure vessels designed to the ASME Boiler and Pressure Vessel Code is remote, the design of nuclear facilities does not provide protection against reactor vessel failure. However, as plants accumulate more and more service time, neutron irradiation reduces the material fracture toughness and initial safety margins.

Results from reactor vessel surveillance programs indicate that up to approximately 20 operating PWRs will have beltline materials with marginal toughness, relative to the requirements of Appendices G and H of 10 CFR Part 20, after comparatively short (approximately 10 EFPY) periods of operation. For most plants now in the licensing process, current criteria, together with the materials currently employed, are adequate to ensure suitable safety margins for reactor vessels throughout their design lives. However, a few plants under licensing review have reactor vessels that have been identified as having the potential for marginal fracture toughness within their design lives; these vessels will have to be reevaluated in the light of the new criteria for long term acceptability.

• RES INTERFACE INFORMATION

Fracture test data was provided by RES and incorporated into NUREG-0744; RES has reviewed NUREG-0744.

• ACRS INTERFACE INFORMATION

Copies of NUREG-0744, Rev. 1, have been forwarded to the ACRS.

• TECHNICAL ASSISTANCE CONTRACTS

NONE

FIN NO	CONTRACTOR	OBLIGATED	EXPENDED
B6486	W.U.T.A.	$103,000	$ 83,088
B0737	ORNL	$415,000	$331,375
A6419	INEL	$100,000	$ 89,740
B6809	NRL	$ 95,650	$ 95,650

• POTENTIAL PROBLEMS

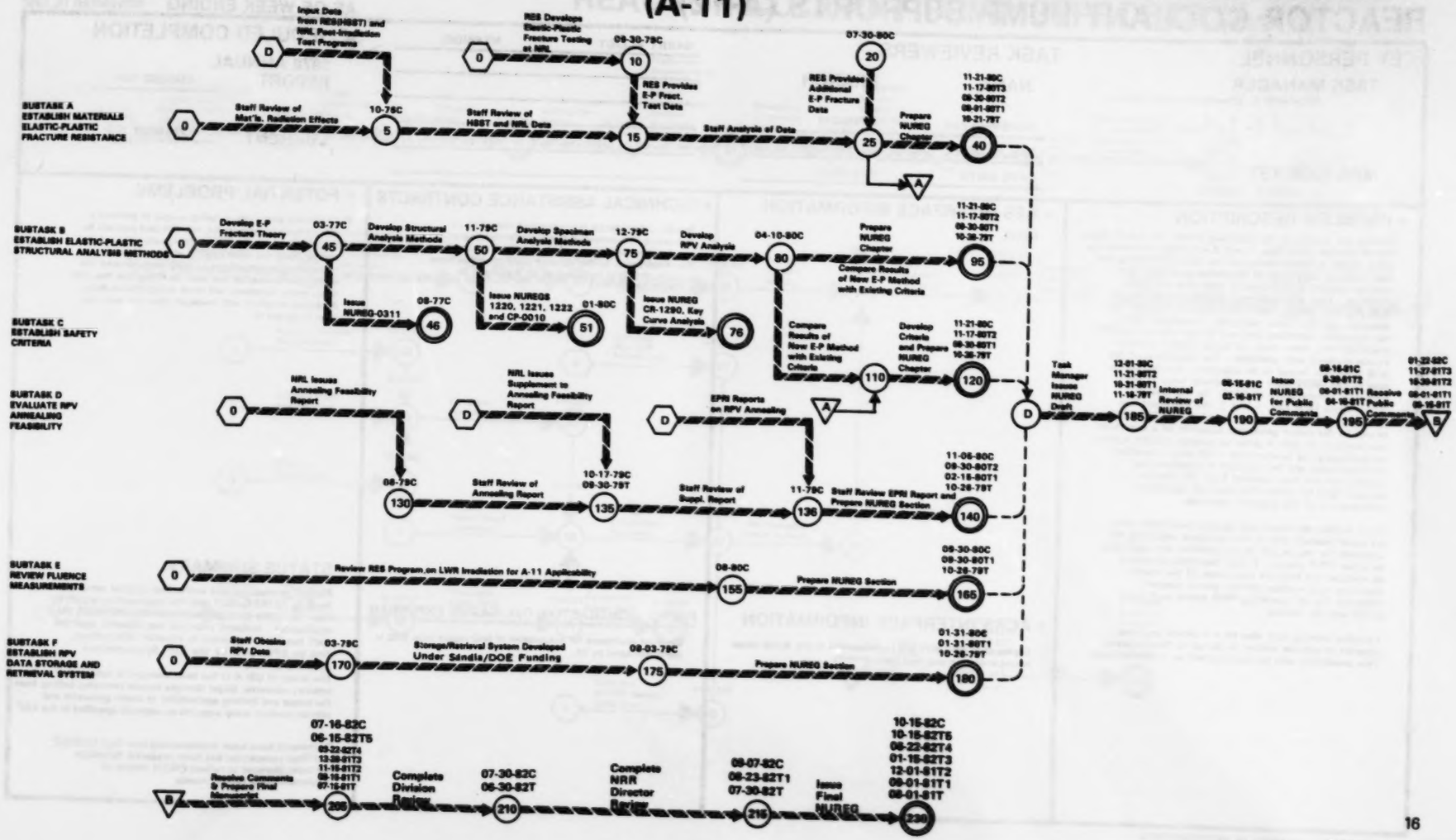
NONE

• STATUS SUMMARY

NUREG-0744, Rev. 1, was printed and issued, which completes Task A-11.

• INDICATES A CHANGE IN INFORMATION

REACTOR VESSEL MATERIALS TOUGHNESS (A-11)



FRACTURE TOUGHNESS OF STEAM GENERATOR AND REACTOR COOLANT PUMP SUPPORTS (A-12)

AS OF WEEK ENDING NOVEMBER 19, 1982

KEY PERSONNEL

TASK MANAGER
RICHARD E. JOHNSON (GIB) X24715
Richard E. Johnson

MPA ANALYST
L. LANIER X28102

TASK REVIEWERS

NAME	BRANCH
DAVE SELLERS	MTEB(DE)
NEIL RANDALL	MTEB(DE)
DAVE SMITH	MTEB(DE)
BARRY ELLIOT	MTEB(DE)

SCHEDULED COMPLETION

1978 ANNUAL REPORT — AUGUST 1979

CURRENT — #08-06-83

• PROBLEM DESCRIPTION

During the course of the licensing action for North Anna Power Station Unit Nos. 1 and 2 a number of questions were raised as to the potential for lamellar tearing and low fracture toughness of the steam generator and reactor coolant pump support materials for those facilities. Two different steel specifications (ASTM A36-70a and ASTM A572-70a) covered most of the material used for these supports. Toughness tests, not originally specified and not in the relevant ASTM specifications, were made on those heats for which excess material was available. The toughness of the A36 steel was found to be adequate, but the toughness of the A572 steel was relatively poor at an operating temperature of 80°F. In the case of North Anna Unit Nos. 1 and 2, the applicant has agreed to raise the temperature of the ASTM A572 beams in the steam generator supports to a minimum temperature of 225°F prior to reactor coolant system pressurization to levels above 1000 psig. Auxiliary electrical heat will be supplied as necessary to supplement the heat derived from the reactor coolant loop to obtain the required operating temperature of the support materials.

Since similar materials and designs have been used on other nuclear plants, the concerns regarding the supports for the North Anna facilities are applicable to other PWR plants. It was therefore necessary to reassess the fracture toughness of the steam generator and reactor coolant pump support materials for all operating PWR plants and those in CP and OL review.

Lamellar tearing may also be a problem in those support structures similar in design to North Anna. This possibility will be investigated on a generic basis.

• RES INTERFACE INFORMATION

NONE

• ACRS INTERFACE INFORMATION

Copies of NUREG-0577 will be forwarded to the ACRS when issued in revised and final form.

• TECHNICAL ASSISTANCE CONTRACTS

Sandia Laboratories has provided expertise in evaluating the potential for lamellar tearing and low fracture toughness of the support materials. The final input to the generic resolution has been submitted and is included as an appendix to NUREG-0577.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED

Contract assistance for preparation of final report from FIRL is being managed by DL.

• POTENTIAL PROBLEMS

Contractor assistance is being sought to perform a risk analysis to support the value/impact analysis for the proposed resolution. The previously selected contractor could not meet conflict of interest requirements as determined by NRC management. An alternative contractor has been identified but the additional work necessary will result in significant delay. A revised schedule will be issued after the new contract is agreed to.

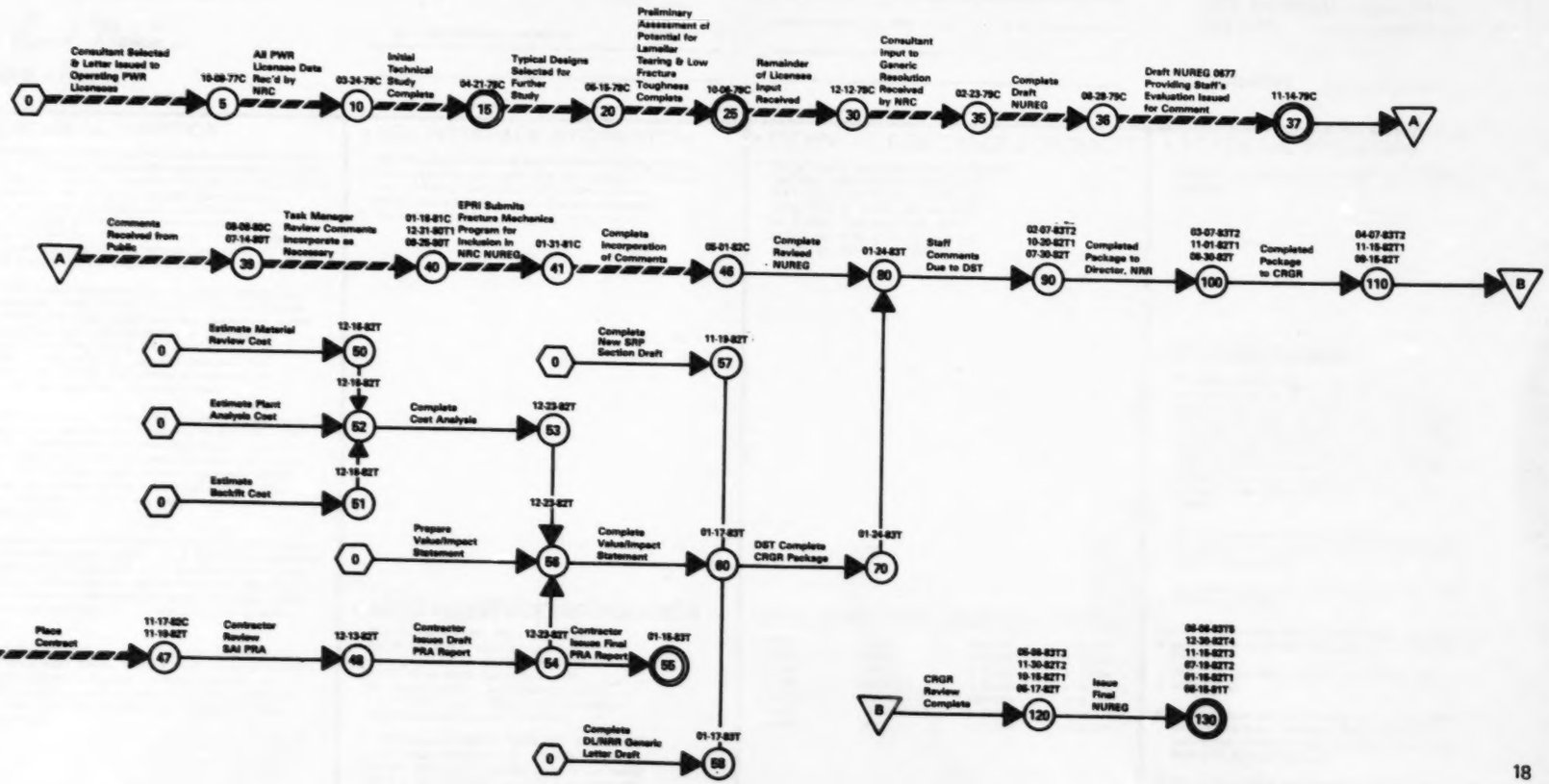
• STATUS SUMMARY

Review procedures and evaluation criteria were set forth in NUREG-0577 and the augmenting letters of May 19, 1980 (to licensees), and May 20, 1980 (to applicants). Industry response was adverse, and the staff has been requested to consider alternatives, first by EPRI and by the AIF Subcommittee on Material Requirements.

The scope of USI A-12 has been changed to reflect staff and industry concerns. Major changes include removing bolting from the scope and limiting applicability to steam generator and primary coolant pump supports as originally specified in the TAP

Comments have been incorporated into final NUREG and final manuscript has been prepared. Schedule has been changed to reflect CRGR review of implementation requirements.

FRACTURE TOUGHNESS OF STEAM GENERATOR AND REACTOR COOLANT PUMP SUPPORTS (A-12)



SYSTEMS INTERACTIONS IN NUCLEAR POWER PLANTS (A-17)

AS OF WEEK ENDING NOVEMBER 19, 1982

KEY PERSONNEL

TASK MANAGER
PAUL NORIAN (Acting) X27487

Paul Norian

MPA ANALYST
L. LANIER X28102

TASK REVIEWERS

NAME	BRANCH
NO ASSIGNED REVIEWERS	

SCHEDULED COMPLETION

1978 ANNUAL REPORT Phase I - 09-79

CURRENT N/A

• PROBLEM DESCRIPTION

The design of a nuclear power plant is accomplished by groups of engineers and scientists organized into engineering disciplines and into scientific disciplines. The reviews performed by the designers include interdisciplinary reviews to assure the functional compatibility of the plant structures, systems, and components. Safety reviews and accident analyses provide further assurance that system functional requirements will be met. These reviews include failure mode analyses.

The NRC review and evaluation of safety systems is accomplished in accordance with the Standard Review Plan (SRP) which assigns primary and secondary review responsibilities to organizational units arranged by plant systems or by disciplines. Each element of the SRP is assigned to an organizational unit for primary responsibility and, where appropriate, to other units for secondary responsibilities.

Thus, the design and analyses by the plant designers, and the subsequent review and evaluation by the NRC staff take into consideration the interdisciplinary areas of concern and account for systems interaction to a large extent. Furthermore, many of our regulatory criteria are aimed at controlling the risks from systems interactions. Examples include the single failure criterion and separation criteria.

Nevertheless, there is some question regarding the interaction of various plant systems, both as to the supporting roles such systems play and as to the effect one system can have on other systems, particularly with regard to whether actions or consequences could adversely affect the presumed redundancy and independence of safety systems.

The problem to be resolved by this task is to identify where the present design, analysis, and review procedures may not acceptably account for potentially adverse systems interaction and to recommend the regulatory action that should be taken to rectify deficiencies in the procedures.

• RES INTERFACE INFORMATION

The Division of Risk Analysis has been consulted during the development and execution of this plan. The Division of Engineering Technology will participate in the development of a Regulatory Guide for Systems Interaction Analyses.

• ACRS INTERFACE INFORMATION

The review is under the cognizance of the ACRS Subcommittee on Plant Arrangements. A meeting was held 02-20-80 in Washington, D.C. with the Subcommittee which recommended a demonstration of Phase I methodology before proceeding much further.

A meeting was held 02-26-82 with the ACRS Subcommittee on Indian Point 3 to describe the PASNY systems interaction program on Indian Point 3. On 03-09-82, the ACRS documented their judgement that the PASNY plan was responsive.

• TECHNICAL ASSISTANCE CONTRACTS

Phase I work was performed by Sandia National Laboratories under FIN # A-1113(9) by NRR for $219K and under FIN # A-1084(8) by OSD for $320K. Expenditures were $197K for FIN # A-1113(9) and $320K for FIN # A-1084(8).

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
# A0403	LLNL	$200,000	$158,000
A3365	BNL	$ 15,000	$ 15,000
B2329	PNL	$ 10,000	$ 10,000
A1296	SANDIA	$ 27,000	$ 16,000
# A-0406	LLNL	$283,000	$196,000
A-3366	BNC	$ 80,000	$ 80,000
B-2339	PNL	$ 80,000	$ 80,000

• POTENTIAL PROBLEMS

LWRs for implementation of interim guidance have not yet been selected.

• STATUS SUMMARY

The anticipated Phase II program will not be pursued as an USI. Work originally planned under TAP A-17 will now be performed under TMI Action Plan Item II-C-3, Systems Interaction. This activity will now be managed by the Systems Interaction Section, Reliability and Risk Assessment Branch, DST. The Section Leader is F. Coffman, X27487. The status of this item is reported in the quarterly updates of the TMI Action Plan Tracking System (APTS).

The Chairman was briefed on 06/25/82 concerning the status of the systems interaction program and how it might proceed.

Stop work orders were issued on contracts A-3366, B-2329, and A-1296.

The documentation of a matrix-digraph method is # expected to be issued 12/82, under contract A-0403.

The initial guidance document is expected to be # issued 01/83.

The staff requested by letter dated June 25, 1982 to Consumer's Power Company that it provide a description of the systems interaction program on Midland 2 for staff evaluation.

The staff's audit of the PASNY systems interaction study on IP3 is proceeding on schedule with technical assistance on A-0406. The contractor's report is expected by August 1983.

# INDICATES A CHANGE IN INFORMATION

DETERMINATION OF SAFETY RELIEF VALVE (SRV) POOL DYNAMIC LOADS AND TEMPERATURE LIMITS FOR BWR CONTAINMENT (A-39)

AS OF WEEK ENDING NOVEMBER 19, 1982

KEY PERSONNEL

TASK MANAGER
T.M. SU (GIB) X27477

MPA ANALYST
L. LANIER X28102

TASK REVIEWERS

NAME	BRANCH

SCHEDULED COMPLETION

1978 ANNUAL REPORT — October 1979

CURRENT
MARK I - 02-29-80C
MARK II - 09-30-82C
MARK III - 10-14-82C

• PROBLEM DESCRIPTION

Operation of BWR primary system pressure relief valves can result in hydrodynamic loads on the suppression pool retaining structures or structures located within the pool. These loads result from:

1. Initial vent clearing of a relief valve pipe; and
2. Steam quenching due to locally high pool temperature.

• RES INTERFACE INFORMATION

NONE

• ACRS INTERFACE INFORMATION

Discussions with the ACRS Subcommittee on Mark III SRV loads and Kuoshong inplant test results was presented on September 25, 1981.

• TECHNICAL ASSISTANCE CONTRACTS

BNL has completed Mark II T-Quencher and Mark III X-Quencher review and is preparing input to the NUREG report.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
A3088	BNL	FY 79 $65,000 FY 80 $50,000	
A3321	BNL	FY 80 $35,000 FY 81 $70,000	

• POTENTIAL PROBLEMS

NONE

• STATUS SUMMARY

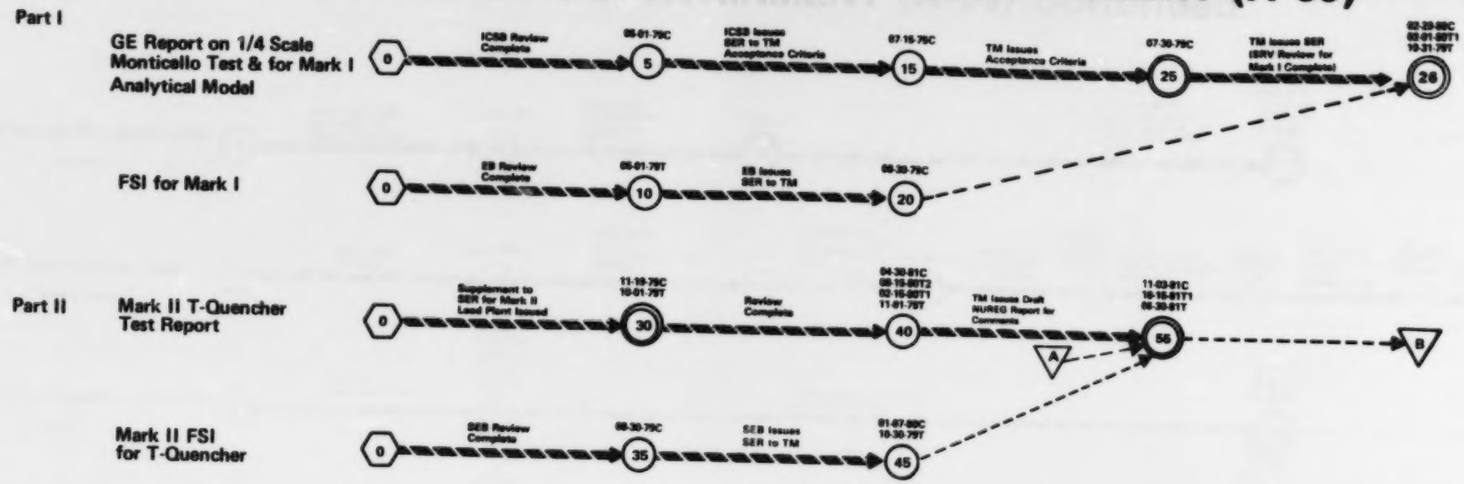
USI A-39 was reviewed by the CRGR on August 11, 1982 and recommended for approval.

NUREGS-0763, -0783, and -0802 were approved for issuance by the EDO on October 14, 1982.

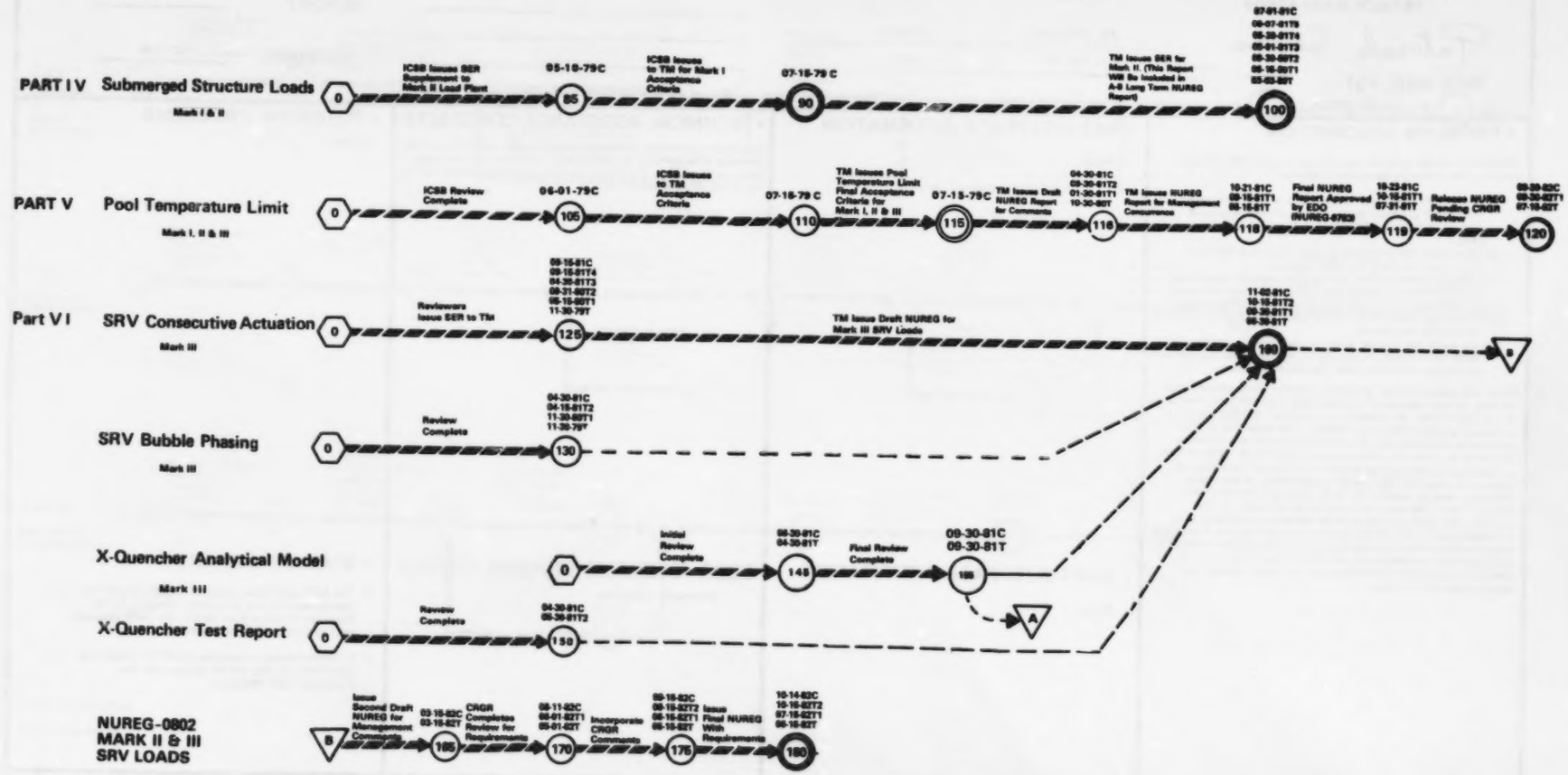
USI A-39 is now complete.

# INDICATES A CHANGE IN INFORMATION

DETERMINATION OF SAFETY RELIEF VALVE (SRV) POOL DYNAMIC LOADS AND TEMPERATURE LIMITS FOR BWR CONTAINMENT (A-39)



DETERMINATION OF SAFETY RELIEF VALVE (SRV) POOL DYNAMICS LOADS AND TEMPERATURE LIMITS FOR BWR CONTAINMENT (A-39) Continued



SEISMIC DESIGN CRITERIA - SHORT TERM PROGRAM (A-40)

AS OF WEEK ENDING NOVEMBER 19, 1982

KEY PERSONNEL

TASK MANAGER
PATRICK SEARS X24216

Patrick Sears

MPA ANALYST
#L. LANIER X28102

TASK REVIEWERS

NAME	BRANCH
#N. CHOKSHI	SEPB/DL
L. REITER	GB/DE
G. BAGCHI	EQB/DE
#K. HERRING	SEPB/DL

SCHEDULED COMPLETION

1978 ANNUAL REPORT — PHASE I – 1979; PHASE II – 1981

CURRENT — #03-16-84

• PROBLEM DESCRIPTION

The seismic design process required by current NRC criteria includes the following sequence of events:

(a) Define the magnitude or intensity of the earthquake which will produce the maximum vibratory ground motion at the site (the safe shutdown earthquake or SSE).
(b) Determine the free-field ground motion at the site that would result if the SSE occurred.
(c) Determine the motion of site structures by modifying the free-field motion to account for the interaction of the site structures with the underlying foundation soil.
(d) Determine the motion of the plant equipment supported by the site structures.
(e) Compare the seismic loads, in appropriate combination with other loads, on structures, systems, and components important to safety, with the allowable loads.

While this seismic design sequence includes many conservative factors, certain aspects of the sequence may not be conservative for all plant sites. At present, it is believed that the overall sequence is adequately conservative. The objective of this program is to investigate selected areas of the seismic design sequence to determine their conservatism for all types of sites, to investigate alternate approaches to parts of the design sequence, to quantify the overall conservatism of the design sequence, and to modify the NRC criteria in the Standard Review Plan if changes are found to be justified. In this manner, this program will provide additional assurance that the health and safety of the public is protected, and if possible, reduce costly design conservatisms by improving (1) current seismic design requirements, (2) NRR's capability to quantitatively assess the overall adequacy of seismic design for nuclear plants in general.

• RES INTERFACE INFORMATION

None.

• ACRS INTERFACE INFORMATION

None.

• TECHNICAL ASSISTANCE CONTRACTS

Lawrence Livermore Laboratory (LLL), under contract to RES, reviewed all reports by 04-30-79. LLL report on recommendations for changes to the seismic design criteria was completed on 12-28-79. (NUREG/CR-1161).

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED

Contracts complete.

• POTENTIAL PROBLEMS

None.

• STATUS SUMMARY

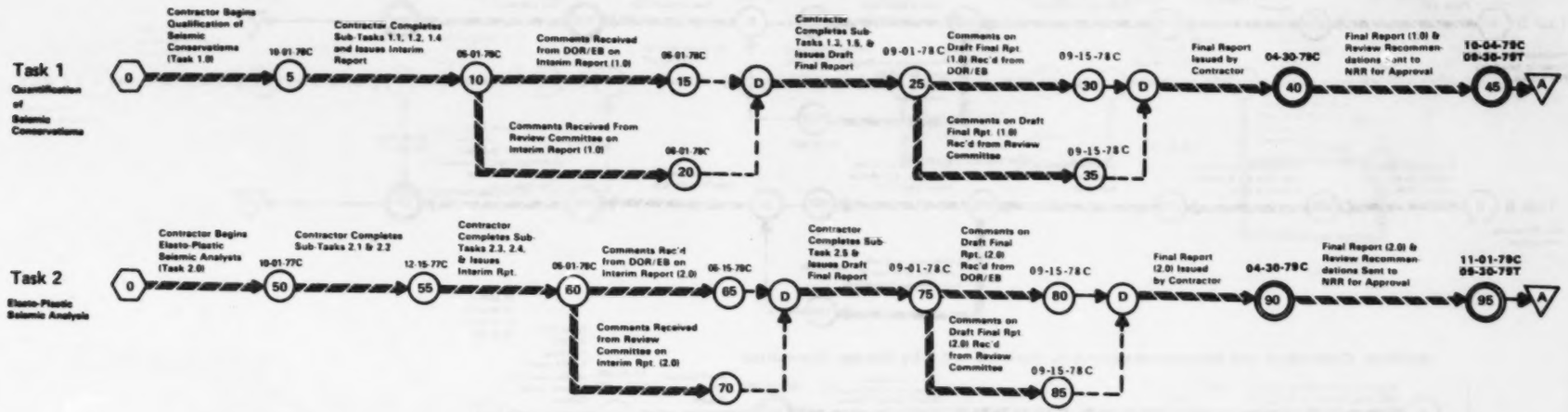
The NRC staff has reached an agreement on acceptance of the proposed changes to the Standard Review Plan (SRP). The SRP changes are currently being prepared.

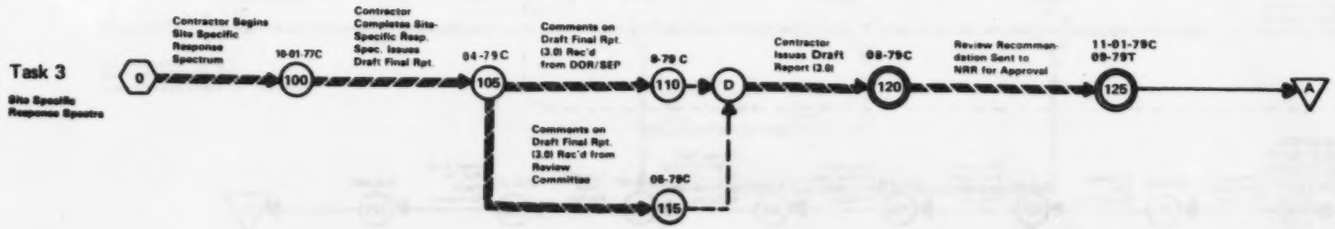
A Value/Impact assessment will be made and submitted to the CRGR along with the proposed SRP changes.

#INDICATES A CHANGE IN INFORMATION

SEISMIC DESIGN CRITERIA - SHORT TERM PROGRAM (A-40)

Phase 1



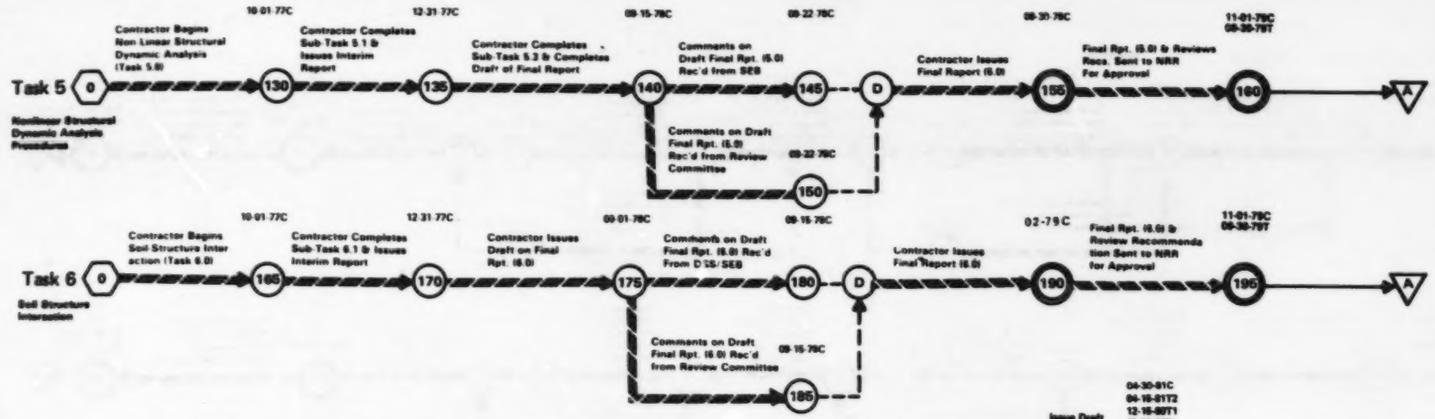


Task 4 Deleted

Seismic After Shocks

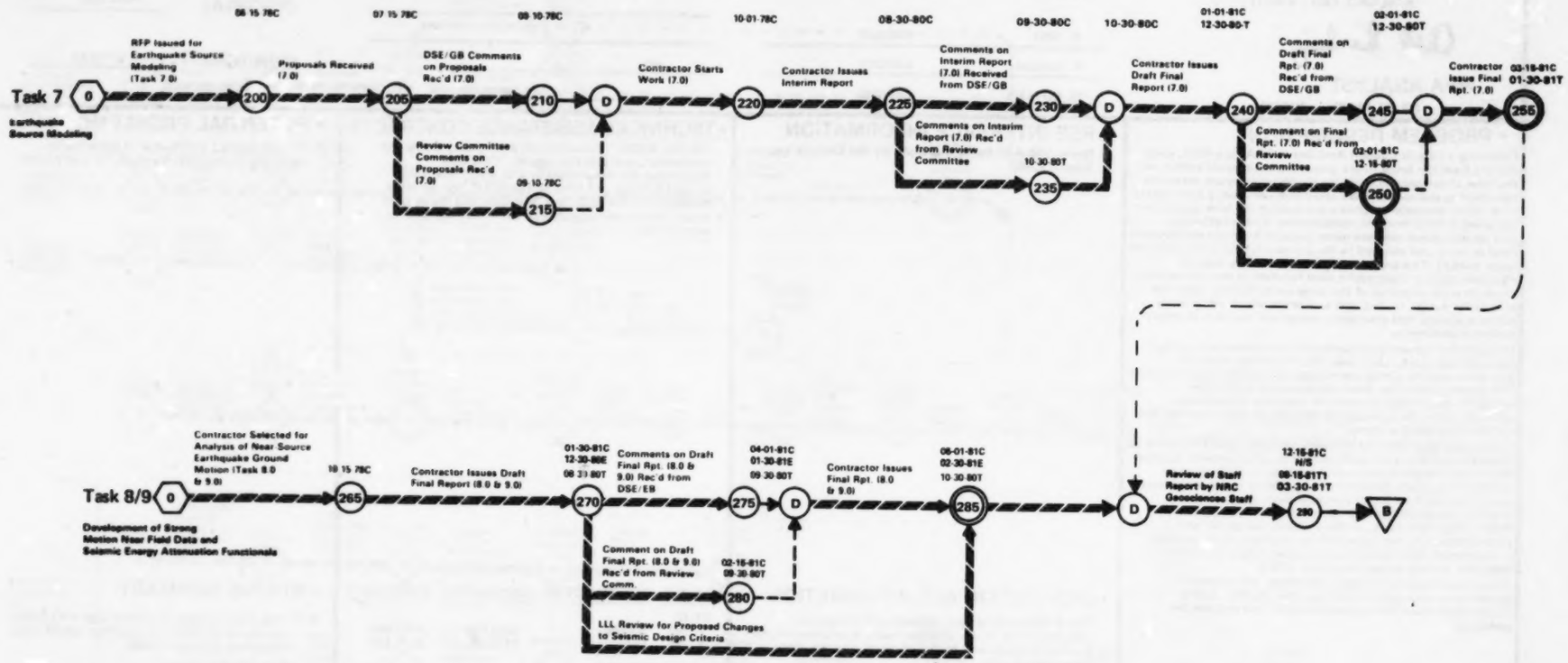
SEISMIC DESIGN CRITERIA - SHORT TERM PROGRAM (A-40)

(Phase 1 Cont.)



Phase 2

SEISMIC DESIGN CRITERIA - SHORT TERM PROGRAM (A-40)



CONTAINMENT EMERGENCY SUMP PERFORMANCE (A-43)

AS OF WEEK ENDING NOVEMBER 19, 1982

KEY PERSONNEL

TASK MANAGER
A. SERKIZ (DST) X24217

MPA ANALYST
B. ST. MARY X28330

TASK REVIEWERS

NAME	BRANCH
F. ORR	RSB/DSI
P. NORIAN	GIB/DST
W. BUTLER	CSB/DSI

SCHEDULED COMPLETION

ORIGINAL April 1982

CURRENT 07-15-83

• PROBLEM DESCRIPTION

Following a Loss of Coolant Accident (LOCA) in a PWR, water flowing from the break in the primary system would collect on the floor of containment. During the injection mode, water for core cooling and containment spray is drawn from a large supply tank. When the water reached a low level in the tank, pumps are realigned to draw from the containment. This is called the recirculation mode wherein water is drawn from the containment floor or sump and pumped to the primary system or containment spray headers. This program addresses the safety issue of adequate sump or suppression pool function in the recirculation mode. It is the objective of this program to develop improved criteria for design, testing, and evaluation which will provide better assurance that emergency sumps will function to satisfy system requirements.

The principal concerns are somewhat interrelated but are best discussed separately. One deals with the various kinds of insulation used on piping and components inside of containment. The concern being that break-initiated debris from the insulation could cause blockage of the sump or otherwise adversely affect the operation of the pumps, spray nozzles, and valves of the safety systems.

The second deals with the hydraulic performance of the sump as related to the hydraulic performance to safety systems supplied therefrom. Preoperational tests have been performed on a number of plants to demonstrate operability in the recirculation mode. Adverse flow conditions have been encountered requiring design and procedural modifications to eliminate them. These conditions, air entrainment, cavitation, and vortex formation, are aggravated by blockage. If not avoided or suppressed, they could result in pump failure during the long term cooling phase following a LOCA.

The concerns relative to debris, blockage, and hydraulic performance also apply to boiling water reactors during recirculation from the suppression pools, and will also be addressed.

• RES INTERFACE INFORMATION

None; USI A-43 being managed by the Generic Issues Branch (GIB).

• ACRS INTERFACE INFORMATION

The ACRS staff is kept abreast of findings and developments on an informal basis. A briefing on the current findings and status of resolution was held December 3, 1981.

An ACRS briefing on A-43 resolution and the proposed revision to R.G. 1.82 is being scheduled in December 1982.

• TECHNICAL ASSISTANCE CONTRACTS

FIN No. A1237, "Containment Emergency Sump Performance", and FIN No. A1298, "Technical Assistance for Resolution of USI A-43", are being funded by RES and NRR respectively. This work is managed by the GIB Task Manager and these combined efforts are expected to be concluded in early FY 83.

	FIN NO.	CONTRACTOR		OBLIGATED	EXPENDED
	FY 81:				
	A1237	Sandia		$225,000	$101,000
	A1298	Sandia		$120,000	$ 75,100
	FY 82:				
	A1237	Sandia	Qtr 1	$334,000	$103,000
			Qtr 2	$315,000	$125,000
			Qtr 3	$315,000	$245,000
#			Qtr 4	$327,000	$327,000
	A1298	Sandia	Qtr 1	$ 71,900	$ 36,000
#			Qtr 2	$221,900	$119,000
#			Qtr 3	$351,900	$126,000
#			Qtr 4	$351,900	$291,000

• POTENTIAL PROBLEMS

Unanticipated additional requirements resulting from CRGR submittal and review.

• STATUS SUMMARY

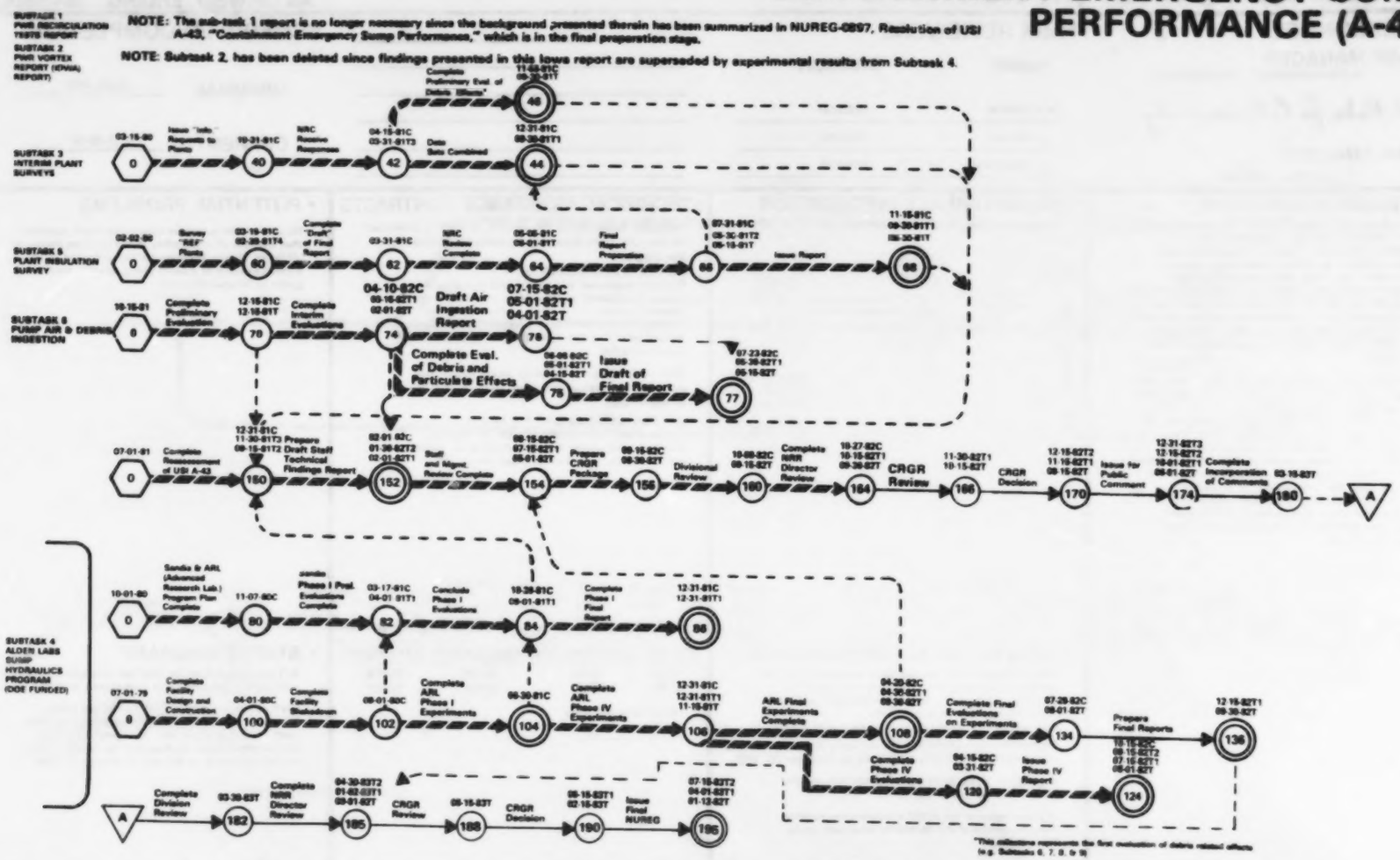
1) Proposed actions to resolve USI A-43 were submitted to CRGR on 10/27/82; CRGR briefing scheduled for 11/24/82.

2) NUREG-0897 (DRAFT) presents technical findings developed for A-43 resolution. Plans are to issue A-43 "For Comment" following CRGR review and approval.

3) All technical support reports (NUREG/CRs) have been issued except ARL's report on insulation debris transport and screen blockage effects. This report is scheduled for issuance by 12/10/82.

# INDICATES A CHANGE IN INFORMATION

CONTAINMENT EMERGENCY SUMP PERFORMANCE (A-43)


SUBTASK 1
PWR RECIRCULATION TESTS REPORT
SUBTASK 2
PWR VORTEX REPORT (IDVM) REPORT)
NOTE: The sub-task 1 report is no longer necessary since the background presented therein has been summarized in NUREG-0897, Resolution of USI A-43, "Containment Emergency Sump Performance," which is in the final preparation stage.
NOTE: Subtask 2, has been deleted since findings presented in this lowa report are superseded by experimental results from Subtask 4.
SUBTASK 3
INTERIM PLANT SURVEYS
SUBTASK 5
PLANT INSULATION SURVEY
SUBTASK 6
PUMP AIR & DEBRIS INGESTION
Complete Preliminary Evaluation of Debris Effects
Issue "Info" Requests to Plants
NRC Receive Responses
Data Bata Combined
Survey "REF" Plants
Complete "Draft" of Final Report
NRC Review Complete
Final Report Preparation
Issue Report
Complete Preliminary Evaluation
Complete Interim Evaluations
04-10-82C
Draft Air Ingestion Report
07-15-82C
Complete Eval. of Debris and Particulate Effects
Issue Draft of Final Report
07-01-81
Complete Assessment of USI A-43
Prepare Draft Staff Technical Findings Report
Staff and Mgmt. Review Complete
Prepare CRGR Package
Divisional Review
Complete NRR Director Review
CRGR Review
CRGR Decision
Issue Public Comment
Complete Incorporation of Comments
A
SUBTASK 4
ALDEN LABS SUMP HYDRAULICS PROGRAM (DOE FUNDED)
Sandia & ARL (Advanced Research Lab.) Program Plan Complete
Complete Phase I Prel. Evaluations Complete
Conclude Phase I Evaluations
Complete Phase I Final Report
Complete Facility Design and Construction
Complete Facility Shakedown
Complete ARL Phase I Experiments
Complete ARL Phase IV Experiments
ARL Final Experiments Complete
Complete Final Evaluations on Experiments
Prepare Final Reports
Complete Phase IV Evaluations
Issue Phase IV Report
Complete Division Review
Complete NRR Director Review
CRGR Review
CRGR Decision
Issue Final NUREG

*This milestone represents the first evaluation of debris related effects for a Subtasks 6, 7, 8, & 9)

STATION BLACKOUT (A-44)

AS OF WEEK ENDING NOVEMBER 19, 1982

KEY PERSONNEL

TASK MANAGER
P. BARANOWSKY (RES) X38880

A. Rubin for P. Baranowsky

MPA ANALYST
L. LANIER X28102

TASK REVIEWERS

NAME	BRANCH
# A. RUBIN	GIB/DST
L. ENGLE	ORB/DL
O. CHOPRA	PSB/DSI

SCHEDULED COMPLETION

ORIGINAL JUNE 1982

CURRENT # 12-15-83

• PROBLEM DESCRIPTION

Electric power for safety systems at nuclear power plants is supplied by two redundant and independent divisions. Each of these electrical divisions includes an offsite alternating current (A.C.) source, an onsite A.C. source (usually diesel-generators), and a direct current (D.C.) source. Appendix A to 10 CFR 50 defines a total loss of offsite power as an anticipated occurrence, and as such, it is required that an independent emergency onsite power supply be provided at nuclear power plants.

The unlikely, but possible loss of A.C. power (that is, the loss of A.C. power from the offsite source and from the onsite source) is referred to as a station blackout. In the event of a station blackout, the capability to cool the reactor core would be dependent on the availability of systems which do not require A.C. power supplies, and on the ability to restore A.C. power in a timely manner. The concern is that the occurrence of a station blackout may be a relatively high probability event and that the consequences of this event may be unacceptable, for example, severe core damage may result.

• RES INTERFACE INFORMATION

RES is managing the Task for NRR

• ACRS INTERFACE INFORMATION

Station Blackout is related to a number of ACRS concerns regarding the reliability of power systems. This task will be coordinated with the committee as the task progresses.

Task action plan outline was discussed with AC/DC Power Reliability Subcommittee on January 22, 1981. Another subcommittee briefing was held on October 30, 1981.

A program status briefing was given to this subcommittee on March 30, 1982.

A presentation to the ACRS subcommittee summarizing the results of the contractor work was made Sept. 8, 1982.

• TECHNICAL ASSISTANCE CONTRACTS

Technical Assistance in the amount of $50K is estimated for this task. The following elements are included in this program:

ORNL FIN B0744 $600K –
Evaluate expected frequency and duration of offsite (preferred) power losses at nuclear power plants.
Estimate the reliability and evaluate the dominant factors affecting the reliability of emergency A.C. power supplies.

$50K in funding was recently added to perform statistical correlation and trend analysis of diesel generator data.

SNL FIN A1302 $250K –
Evaluate the risks posed by station blackout accidents and assess the effectiveness of safety improvements in reducing those risks.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED*
B0744	ORNL	$600K	$580K
A1302	SNL	$250K	$140K

*As of June 30, 1982

• POTENTIAL PROBLEMS

Results of the extensive contractor work indicate that because of availability in the systems design at different plants, implementation will need to be on a plant specific or a plant group specific basis in order to be cost effective.

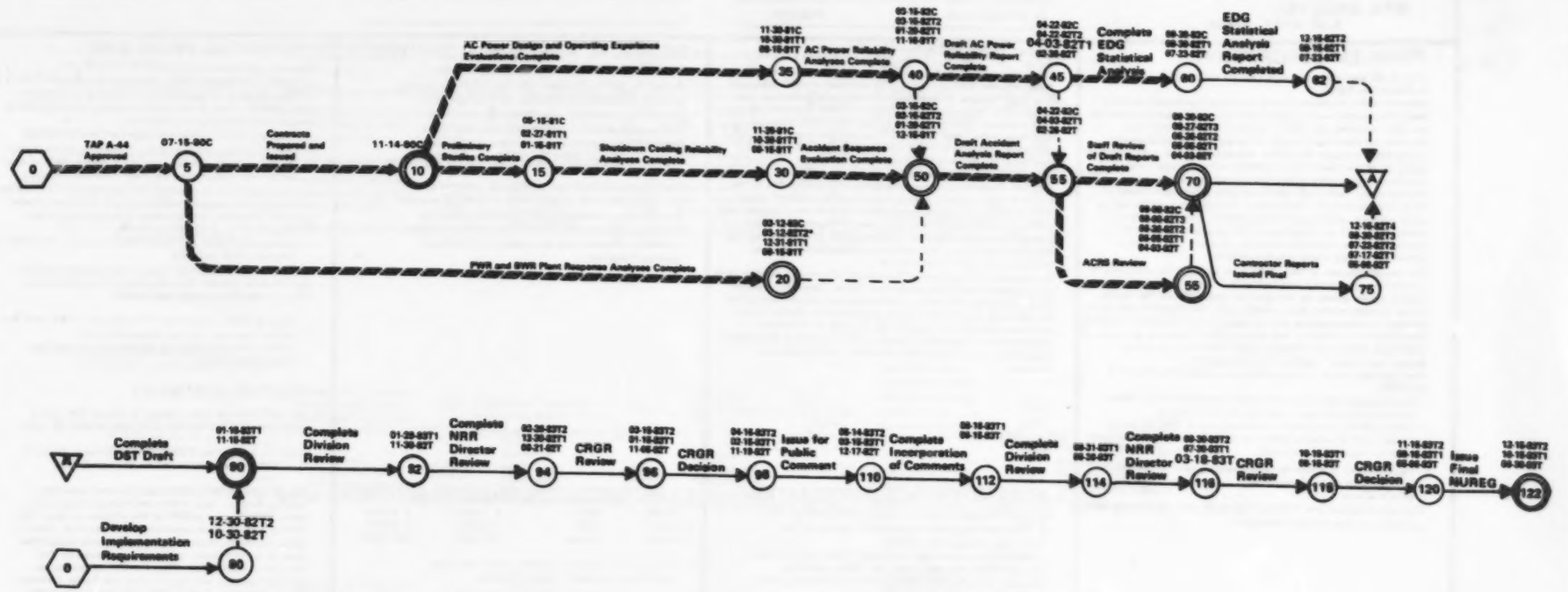
• STATUS SUMMARY

A Task Action Plan for this task was issued in July 1980 and work is processing in accordance with this plan.

Draft contractor reports from Sandia and ORNL have been received and were reviewed by the NRC staff. Final reports are now in preparation. A contractor presentation summarizing the results of the studies was made to the staff on August 11, 1982.

# INDICATES A CHANGE IN INFORMATION

STATION BLACKOUT (A-44)



*Preliminary analyses completed 3-81.

SHUTDOWN DECAY HEAT REMOVAL REQUIREMENTS (A-45)

AS OF WEEK ENDING NOVEMBER 19,1982

KEY PERSONNEL

TASK MANAGER
ANDREW MARCHESE X24712

P. Nowa for A. Marchese

MPA ANALYST
B. ST. MARY X28330

TASK REVIEWERS

NAME	BRANCH
T. MARSH	RSB/DSI
F. ROSA	ICSB/DSI
M. SRINIVASAN	PSB/DSI
E. McPEEK	SSPB/DL
P. WAGNER	ORB-4/DL
M. CUNNINGHAM	DRA/RES

SCHEDULED COMPLETION

ORIGINAL	10/85 PRELIMINARY NETWORK
CURRENT	10-30-85

• PROBLEM DESCRIPTION

Task A-45 was approved as a USI by the NRC in December 1980.
Although many improvements to the steam generator auxiliary feedwater system were required of the reactor manufacturers by the NRC following the TMI-2 accident, the staff feels that providing an alternative means of decay heat removal could substantially increase the plants' capability to deal with a broader spectrum of transients and accidents and potentially could, therefore, significantly reduce the overall risk to the public. Consequently, Task A-45 will investigate alternative means of decay heat removal in PWR plants, including but not limited to using existing equipment where possible. This Unresolved Safety Issue will also investigate the need and possible design requirements for improving reliability of decay heat removal systems in boiling water reactors (BWRs).
The overall purpose of Task A-45 is to evaluate the adequacy of current licensing design requirements, in order to ensure that nuclear power plants do not pose an unacceptable risk due to failure to remove shut-down decay heat. The objective will be to develop a comprehensive and consistent set of shutdown cooling requirements for existing and future LWRs, including the study of alternative means of shutdown decay heat removal and of diverse "dedicated" systems for this purpose.
The approach taken to this problem comprises the following subtasks:
- Development of criteria to judge acceptability of decay heat removal systems in existing & future plants.
- Development of means for improvement of DHR systems.
- Assessment of existing plants to identify those in which DHR systems require improvement.
- Development of a plan for implementing new requirements, if any, for DHR systems required to meet the acceptance criteria above.

The interrelation and relative timing of each of the sub-tasks is shown on the schedule network.

• RES INTERFACE INFORMATION

Close coordination and cooperation will be required on Task A-45 between NRR and RES. RES assistance will be required from the Divisions of Risk Analysis, Accident Evaluation, and Engineering Technology. The Division of Risk Analysis will provide technical input from their Sandia Laboratory Program on Alternate Decay Heat Removal Concepts, technical evaluations relative to reliability and risk assessment for shutdown decay heat removal systems, and input from Task A-44, "Station Blackout," relative to shutdown cooling systems. The Division of Accident Evaluation will provide technical input relative to the transient response of existing and improved shutdown decay heat removal systems to transient events and small LOCAs. This will also include performing (in-house, contractors) detailed thermal-hydraulics analyses where required to support existing and improved decay heat removal systems behavior under transient and accident conditions.
The Division of Engineering Technology will provide assistance in the preparation and publication (i.e., Rules, Regulatory Guides, Standard Review Plans) of a consistent and comprehensive set of shutdown decay heat removal requirements. At the outset, particular attention will need to be focused on the further development of Regulatory Guide 1.139, "Guidance for Residual Heat Removal to Achieve and Maintain Cold Shutdown," Revision 1.

• ACRS INTERFACE INFORMATION

- Task Manager briefed the full committee on July 9, 1982 on the present plans and content of the A-45 Task Action Plan (TAP).
- ACRS Subcommittee on Decay Heat Removal Systems (DHRS) met on March 16, 1982, and Task Manager provided the subcommittee with an update of the TAP.
- Further meetings with the full committee and subcommittee on DHRS will be held as the work on A-45 progresses and certain pre-determined milestones are completed.

• TECHNICAL ASSISTANCE CONTRACTS

- Implemented a technical assistance contract on August 5, 1981 with BNL (FIN-A-3381) for $100K to make a determination whether it is feasible to separate all currently operating U.S. commercial LWR's into groups based on whether they have similar DHRS design characteristics to the plants covered in WASH-1400, RSSMAP, and IREP.

Implemented a technical assistance contract on May 10, 1982 with Sandia (FIN A1309) to provide overall project management, technical direction and integration for the entire Task A-45 program, including selection and management of subcontractors if needed.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
B8835	UCLA	$ 10,000	$ 10,000
# A3381	BNL	$100,000	$ 80,000
# A1309	Sandia	$637,000	$529,000*

* Includes the following funding which has been committed to support subcontracting.

UCLA	$168,000
ORNL	$112,000

• POTENTIAL PROBLEMS

Potential problem areas identified to date are:
- Obtaining sufficient NRR manpower to work on Task A-45.
- Development of appropriate reliability or quantitative goals for Task A-45 and translation of probabilistic results into licensing requirements.
- Program funding must be approved and obtained.
- Obtaining necessary operating experience on shutdown cooling systems including the most current DHR system design information resulting from Post-TMI changes.
- Uncertainty in the quality of information that will be available from ongoing reliability and risk assessments, on what schedule, and whether information can be extrapolated to all operating plants.
- Close coordination and cooperation needed within NRR and between NRR and RES.
- Number of plants that need to be assessed may be significantly greater than the plants that will have a PRA or reliability study performed.
- Time to obtain approval of funding by NRR and the Senior Contract Review Board has been greater than expected by about six months (See Status Summary).

• STATUS SUMMARY

The task manager was selected in March 1981 and a draft Task Action Plan was published on May 22, 1981.

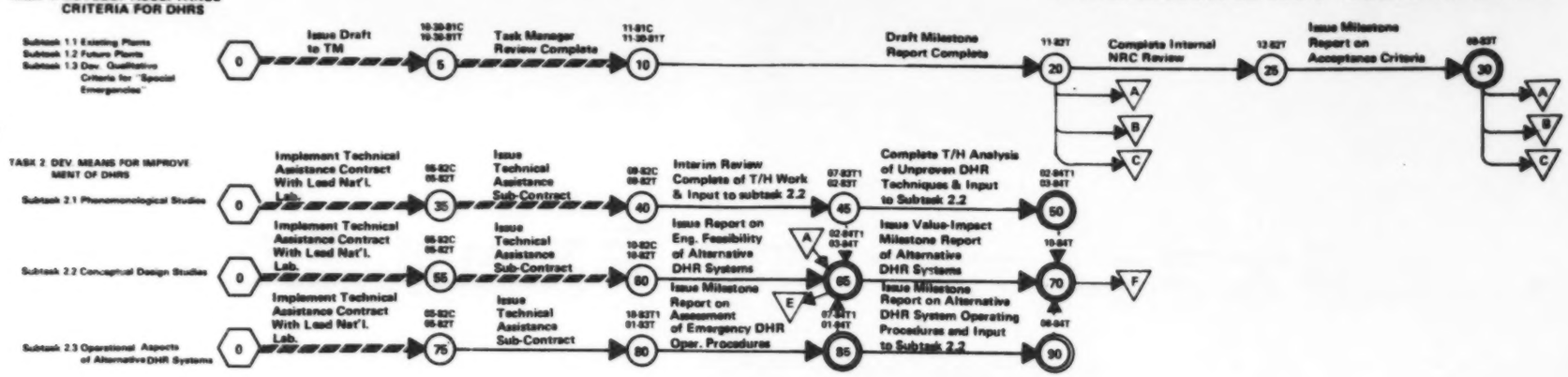
Based on internal staff and ACRS comments, Revision 0 of TAP A-45 was approved on October 7, 1981.

The Director of NRR did not approve the requested funding level for Task A-45 technical assistance work and a "HOLD" was placed on the program in Nov. 1981. Various options were considered for Task A-45, most of which included reduced scope, funding and completion time. A recommended option was approved by The Director of NRR in Feb. 1982. A technical assistance contract package was approved by the Senior Contract Review Board on April 9, 1982 and implemented with the Contractor (Sandia) on May 10, 1982. Revision 1 of TAP A-45 consistent with the above actions was issued on June 2, 1982.

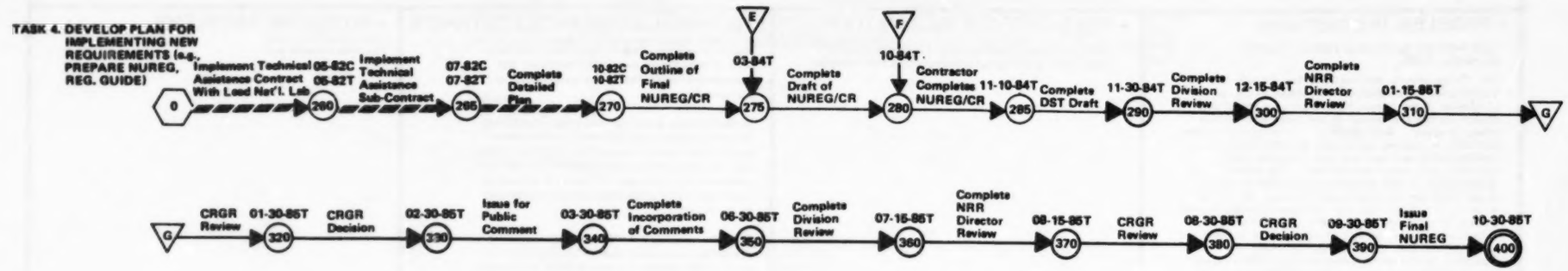
The first interim milestone report on "Quantitative Probabilistic Goal for DHR System" will be issued in draft form for NRC staff review on November 30, 1982.

# INDICATES A CHANGE IN INFORMATION

SHUTDOWN DECAY HEAT REMOVAL REQUIREMENTS (A-45)
PRELIMINARY NETWORK


TASK 1. DEVELOP ACCEPTANCE CRITERIA FOR DHRS
Subtask 1.1 Existing Plants
Subtask 1.2 Future Plants
Subtask 1.3 Dev. Qualitative Criteria for "Special Emergencies"
Issue Draft to TM
Task Manager Review Complete
Draft Milestone Report Complete
Complete Internal NRC Review
Issue Milestone Report on Acceptance Criteria
TASK 2. DEV. MEANS FOR IMPROVEMENT OF DHRS
Subtask 2.1 Phenomenological Studies
Subtask 2.2 Conceptual Design Studies
Subtask 2.3 Operational Aspects of Alternative DHR Systems
Implement Technical Assistance Contract With Lead Nat'l. Lab.
Issue Technical Assistance Sub-Contract
Interim Review Complete of T/H Work & Input to subtask 2.2
Complete T/H Analysis of Unproven DHR Techniques & Input to Subtask 2.2
Issue Report on Eng. Feasibility of Alternative DHR Systems
Issue Value-Impact Milestone Report of Alternative DHR Systems
Issue Milestone Report on Assessment of Emergency DHR Oper. Procedures
Issue Milestone Report on Alternative DHR System Operating Procedures and Input to Subtask 2.2
TASK 3. ASSESS ADEQUACY OF DHRS IN "EXISTING" LWR'S
Subtask 3.2 Assess Adequacy of DHRS in Existing Plants on Probabilistic Basis
Subtask 3.3 Group Other Existing Plants for Assessments of Adequacy of DHRS
Subtask 3.5 Assess Adequacy of DHRS in Existing Plants on Deterministic Basis
Implement Technical Assistance Contract with Lead Nat'l. Lab
Implement Technical Assistance Sub-Contracts
Complete Draft Milestone Report
Issue Milestone Report on LWRS that do not meet Probabilistic Screening Criteria
Complete Discussion with National Labs
Implement Technical Assistance Contract With BNL
Complete Preliminary Grouping of Existing Plants
Issue Interim Final Report on Grouping of Plants
Complete Grouping of Existing Plants as Risk and Reliability Assessments Become Available
Implement Technical Assistance Contract With Lead Nat'l. Lab
Implement Technical Assistance Sub-Contract
Complete Draft Milestone Report
Issue Milestone Report on LWRS that do not meet Deterministic Screening Criteria

SHUTDOWN DECAY HEAT REMOVAL REQUIREMENTS (A-45) (CONTINUED)
PRELIMINARY NETWORK



SEISMIC QUALIFICATION OF EQUIPMENT IN OPERATING PLANTS (A-46)

AS OF WEEK ENDING NOVEMBER 19, 1982

KEY PERSONNEL

TASK MANAGER
PATRICK SEARS X24216
Patrick Sears

MPA ANALYST
B. ST. MARY X28330

TASK REVIEWERS

NAME	BRANCH
JERRY JACKSON	EQB/DE
KENNETH HERRING	SEPB/DL
JOHN KNOX	PSB/DSI

SCHEDULED COMPLETION

ORIGINAL 12-15-83

CURRENT #04-30-84

• PROBLEM DESCRIPTION

Task A-46 was approved as a USI by the NRC December, 1980.

The design criteria and methods for the seismic qualification of mechanical and electrical equipment in nuclear power plants have undergone significant change during the course of the commercial nuclear power program. Consequently, the margins of safety provided in existing equipment to resist seismically induced loads and perform the intended safety functions may vary considerably. The seismic qualification of the equipment in operating plants must, therefore, be reassessed to ensure the ability to bring the plant to a safe shutdown condition when subject to a seismic event. The objective of this Unresolved Safety Issue is to establish an explicit set of guidelines that could be used to judge the adequacy of the seismic qualification of mechanical and electrical equipment at all operating plants in lieu of attempting to backfit current design criteria for new plants. This guidance will concern equipment required to safely shutdown the plant, as well as equipment whose function is not required for safe shutdown, but whose failure could result in adverse conditions which might impair shutdown functions. Also, explicit guidelines will be established for use in requalifying equipment whose seismic qualification was found to be inadequate.

A preliminary breakdown of the tasks is as follows:

Task 1	Develop Basis for Equipment to be Qualified
Task 2	Survey of Equipment Qualification Methods
Task 3	Develop Methods for In-Situ Qualification of Equipment in Operating Plants
Task 4	Seismic Qualification of Equipment Using Non-Nuclear Plant Dynamic Response Information
Task 5	Develop Guidelines for Qualification Response Spectra
Task 6	Document Results of Task A-46 and Prepare a Final NUREG

• RES INTERFACE INFORMATION

RES has placed a contract with Southwest Research Institute to survey seismic qualification methodology.

• ACRS INTERFACE INFORMATION

A preliminary Task Action Plan was presented to the ACRS. The staff plans to meet with the ACRS at appropriate times to present the status of program results.

• TECHNICAL ASSISTANCE CONTRACTS

EG&G is well into Task 3. They have completed 90% of their library search and contacted test laboratories concerning in-situ qualification of equipment.

A plant owners group has collected a significant amount of data on equipment in non-nuclear plants which have undergone earthquakes. Similar data on equipment in nuclear plants has also been collected. LLL, as a part of Task 4, is evaluating the feasibility of using these data for equipment qualification.

BNL has completed work on developing methodology to select equipment to be qualified on a typical PWR. Work on a typical BWR has started.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
#A0423	LLNL	$75K	$65K (est.)
#A6474	INEL	$165K	$135K (est.)
#A3397	BNL	$207K	$194K (est.)

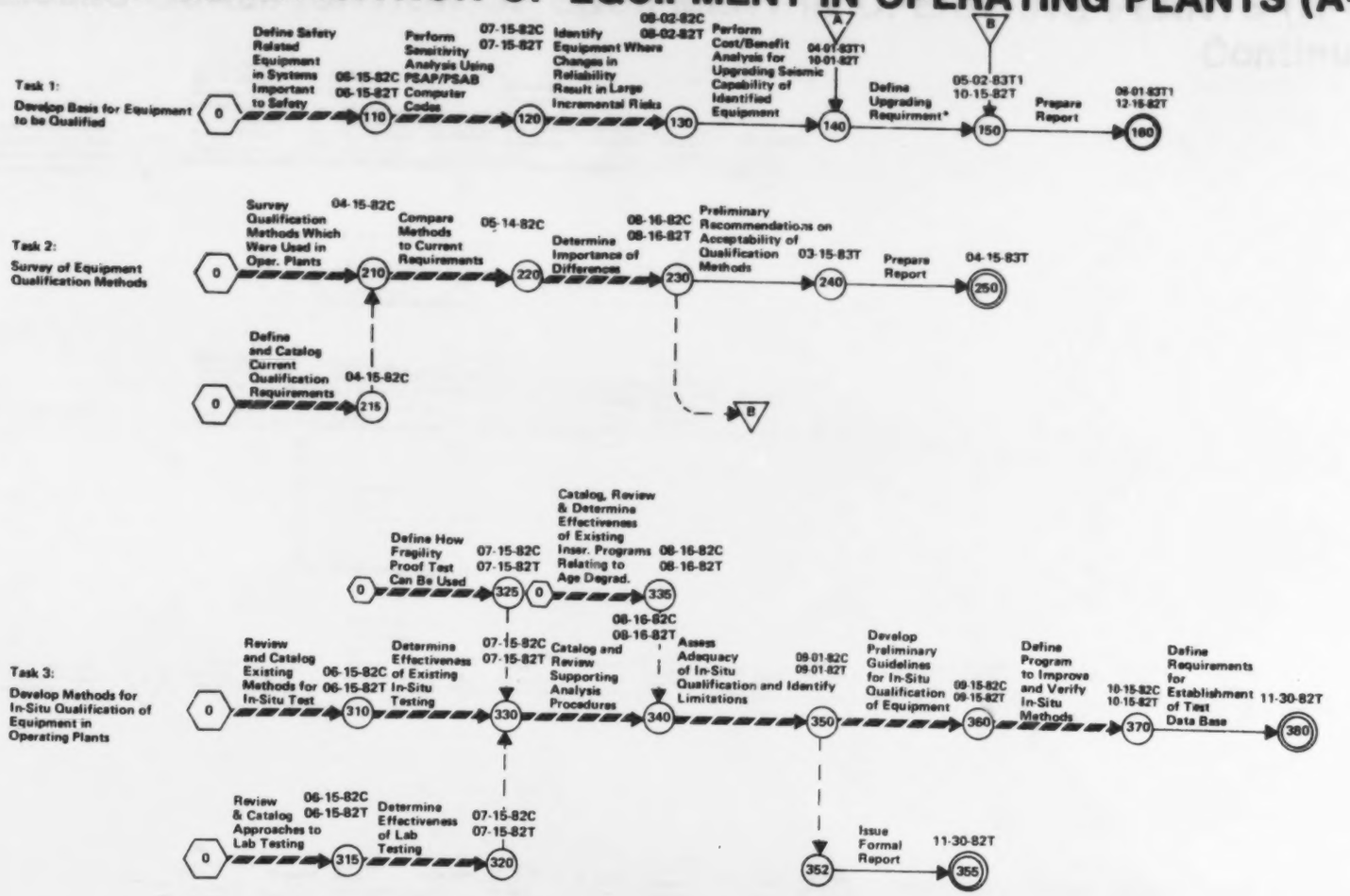
• POTENTIAL PROBLEMS

• STATUS SUMMARY

Work is progressing on all tasks with all contractors well into their subtasks. Meetings have taken place # with members of the Owner's Group who have surveyed nuclear and non-nuclear equipment, and LLNL is evaluating the feasibility of using experience data to qualify equipment.

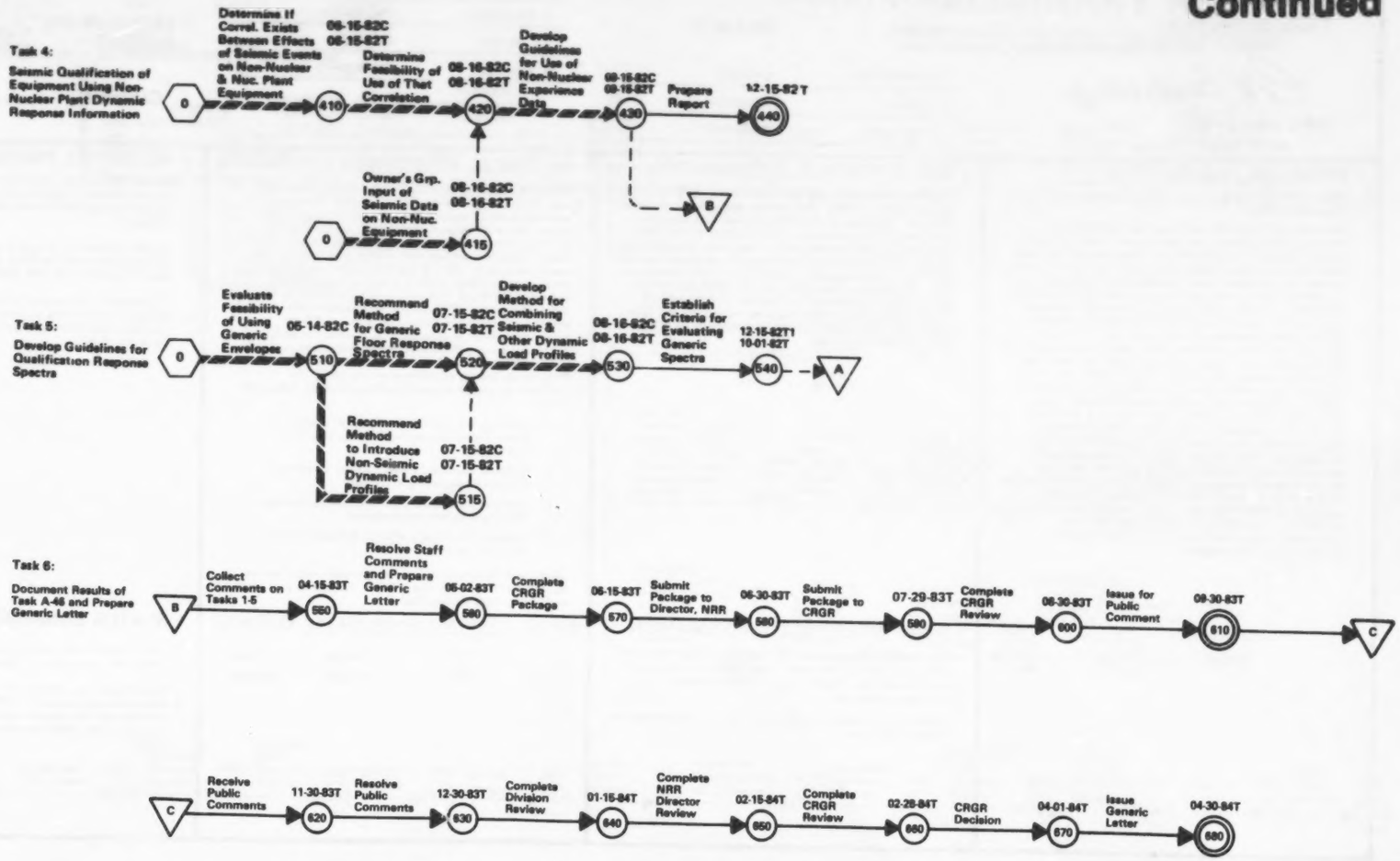
# INDICATES A CHANGE IN INFORMATION

SEISMIC QUALIFICATION OF EQUIPMENT IN OPERATING PLANTS (A-46)



*Required Upgrading: (1) Upgrade None, (2) Upgrade High Risk Only, (3) Upgrade All.

SEISMIC QUALIFICATION OF EQUIPMENT IN OPERATING PLANTS (A-46)
Continued



SAFETY IMPLICATIONS OF CONTROL SYSTEMS (A-47)

AS OF WEEK ENDING NOVEMBER 19, 1982

KEY PERSONNEL

TASK MANAGER
ANDREW SZUKIEWICZ (GIB) X24713

MPA ANALYST
B. ST. MARY X28330

TASK REVIEWERS

NAME	BRANCH
D. BASDEKAS	DFO/RES
C. ROSSI	ICSB/DSI
# E. CHELLIAH	RRAB/DST
F. ORR	RSB/DSI
M. CHIRAMAL	PSU/AEOD
J. T. BEARD	ORAB/DL
W. KENNEDY	PTRB/DHFS

SCHEDULED COMPLETION

1978 ANNUAL REPORT — NOT DETERMINED

CURRENT — N/S

• PROBLEM DESCRIPTION

Task A-47 was approved as a USI by the NRC in December 1980.

This issue concerns the potential for accidents or transients being made more severe as a result of control system failures or malfunctions. These failures or malfunctions may occur independently or as a result of the accident or transient under consideration and would be in addition to any control system failure that may have initiated the event. Although it is generally believed that control system failures are not likely to result in loss of safety functions which could lead to serious events or result in conditions that safety systems are not able to cope with, in-depth studies have not been performed to support this belief. The potential for an accident that would affect a particular control system - and the effects of the control system failures - will differ from plant to plant. Therefore, it is not likely that it will be possible to develop generic answers to these concerns, but rather plant-specific reviews will be required. The purpose of this Unresolved Safety Issue is to define generic criteria that may be used for plant-specific reviews. A specific subtask of this issue will be to study the steam generator overfill transient in PWRs and the reactor overfill transient in BWRs to determine and define the need for preventive and/or mitigating design measures to accommodate this transient.

• RES INTERFACE INFORMATION

Close coordination will be required on Task A-47 between NRR and RES. RES assistance will be required from the Division of Facilities Operations and the Division of Risk Analysis. The Division of Facilities Operations will provide input from their research programs (currently being developed in conjunction with A-47) on control system interactions. The Division of Risk Analysis will provide input on their current ongoing ORNL program ("Evaluation of Pressurized Thermal Shock"). Other research activities are being evaluated to determine their significance to A-47 and may be identified later.

• ACRS INTERFACE INFORMATION

The ACRS Subcommittee on Electrical Power Systems (Dr. Kerr) met on the issue on 1/23/81 to initiate an approximate six month study at the request of Chairman Ahearne.

Elements of conceptual Task Action Plan were discussed with ACRS on 05-07-81.

• TECHNICAL ASSISTANCE CONTRACTS

During the week of July 20, 1981, meetings were held with INEL, EG&G (Idaho Falls), Battelle Northwest and Lawrence Livermore Labs. for the purpose of discussing the A-47 objectives and to assess the type of activities that could best be conducted by these organizations. Similar meetings with ORNL and Sandia Labs. were held July 9, 1981. A Technical Assistance (T.A.) contract with ORNL (FIN B-0467) was established through NRR/RES. RES will conduct a review of two different types of PWR designs, and perform the activities identified in Tasks 1, 2, 3 and 4 of Task Action Plan A-47. A separate T.A. contract to perform the review on one BWR type design was established with INEL (FIN A-6477). A T.A. contract to review a third PWR design will be established in late 1982. The technical assistance contracts will perform the following tasks:

1. Evaluate Control System Failures Leading to Steam Generator and/or Reactor Overfill Transients
2. Evaluate Control System Failures Leading to Reactor Overcooling Transients
3. Evaluate (All Others) Control System Actions That Have Safety Implications
4. Evaluate the Effects of Loss of Power to the Control Systems

Additional Technical Assistance Contract work will be needed for Task A-47 and is being defined as part of the developing Task Action Plan.

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
# A-6477	INEL	$142,000	$ 73,000
# B-0467	ORNL	$600,000	$534,000

• POTENTIAL PROBLEMS

Obtain necessary operating experience information on Non-Safety Control Systems for the Nuclear Steam System (NSS) and the Balance of Plant (BOP) systems.

Obtain necessary design information (i.e., schematics and systems descriptions) for the Non-Safety Control System designs for the NSS and the BOP designs.

Traditionally, licensees were not required to provide design or operating experience on Non-Safety Systems and therefore this information may be difficult to obtain.

• STATUS SUMMARY

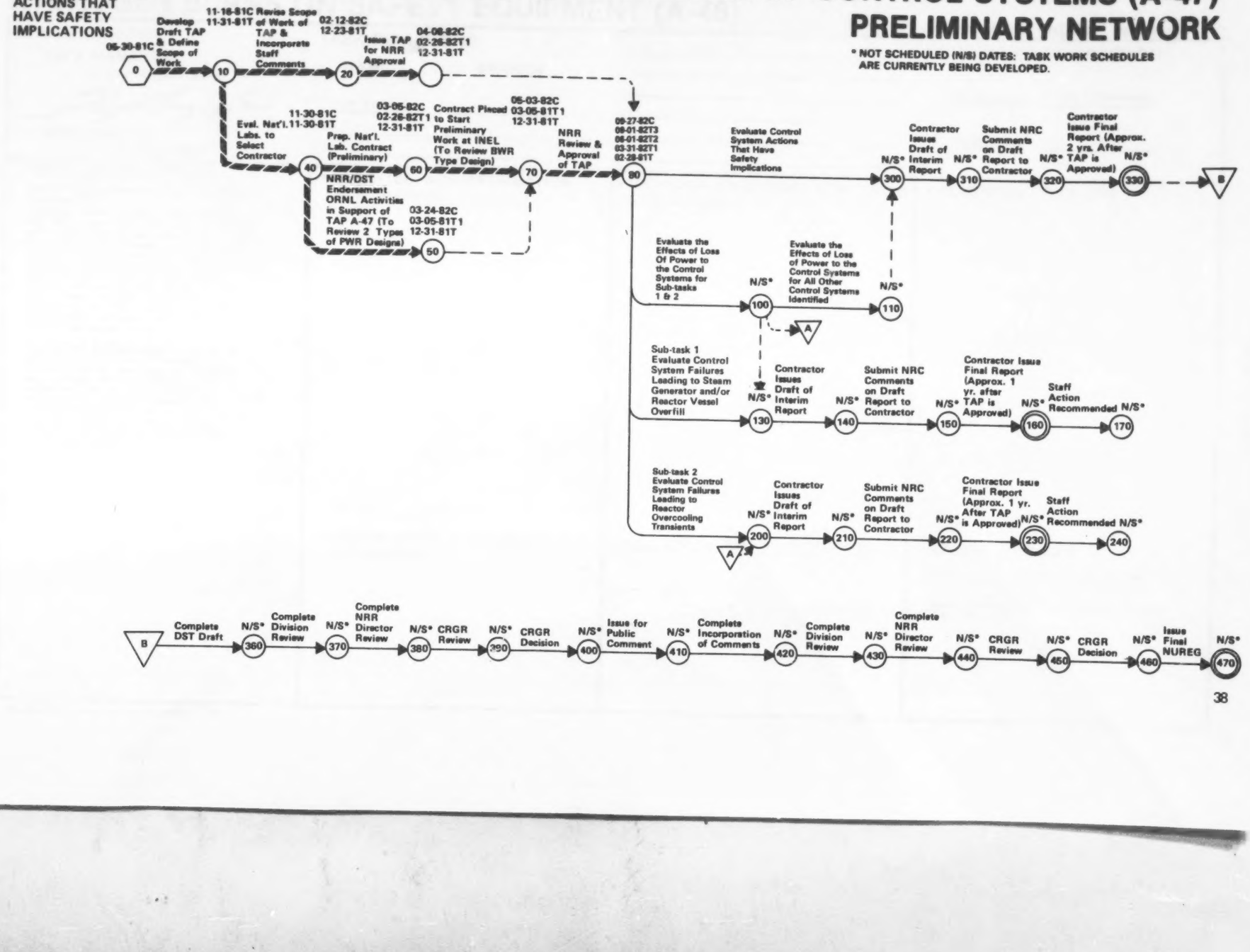
A task manager has been selected. An approved Task Action Plan was issued on September 27, 1982.

A work order was placed with INEL (May 3, 1982) to start preliminary work on a BWR plant review. Model development for computer simulation of the control systems on two PWR-type plants is underway at ORNL.

# INDICATES A CHANGE IN INFORMATION

SAFETY IMPLICATIONS OF CONTROL SYSTEMS (A-47)
PRELIMINARY NETWORK

* NOT SCHEDULED (N/S) DATES: TASK WORK SCHEDULES ARE CURRENTLY BEING DEVELOPED.


EVALUATE CONTROL SYSTEM ACTIONS THAT HAVE SAFETY IMPLICATIONS
05-30-81C
Develop Draft TAP & Define Scope of Work
11-16-81C Revise Scope 11-31-81T of Work of TAP & Incorporate Staff Comments
02-12-82C 12-23-81T
Issue TAP for NRR Approval
04-06-82C 02-26-82T1 12-31-81T
11-30-81C 11-30-81T
Eval. Nat'l. Labs. to Select Contractor
Prep. Nat'l. Lab. Contract (Preliminary)
03-05-82C 02-26-82T1 12-31-81T
Contract Placed to Start Preliminary Work at INEL (To Review BWR Type Design)
05-03-82C 03-05-81T1 12-31-81T
NRR Review & Approval of TAP
09-27-82C 09-01-82T3 09-01-82T2 03-31-82T1 02-28-81T
NRR/DST Endorsement ORNL Activities in Support of TAP A-47 (To Review 2 Types of PWR Designs)
03-24-82C 03-05-81T1 12-31-81T
Evaluate Control System Actions That Have Safety Implications
Contractor Issues Draft of Interim Report
Submit NRC Comments on Draft Report to Contractor
Contractor Issue Final Report (Approx. 2 yrs. After TAP is Approved)
N/S*
Evaluate the Effects of Loss Of Power to the Control Systems for Sub-tasks 1 & 2
Evaluate the Effects of Loss of Power to the Control Systems for All Other Control Systems Identified
Sub-task 1 Evaluate Control System Failures Leading to Steam Generator and/or Reactor Vessel Overfill
Contractor Issues Draft of Interim Report
Submit NRC Comments on Draft Report to Contractor
Contractor Issue Final Report (Approx. 1 yr. after TAP is Approved)
Staff Action Recommended
Sub-task 2 Evaluate Control System Failures Leading to Reactor Overcooling Transients
Contractor Issues Draft of Interim Report
Submit NRC Comments on Draft Report to Contractor
Contractor Issue Final Report (Approx. 1 yr. After TAP is Approved)
Staff Action Recommended
Complete DST Draft
Complete Division Review
Complete NRR Director Review
CRGR Review
CRGR Decision
Issue for Public Comment
Complete Incorporation of Comments
Complete Division Review
Complete NRR Director Review
CRGR Review
CRGR Decision
Issue Final NUREG

HYDROGEN CONTROL MEASURES AND EFFECTS OF HYDROGEN BURNS ON SAFETY EQUIPMENT (A-48)

AS OF WEEK ENDING NOVEMBER 19, 1982

KEY PERSONNEL

TASK MANAGER

#TSUNG MING SU X27477

MPA ANALYST

B. ST. MARY X28330

TASK REVIEWERS

NAME	BRANCH
NOT ASSIGNED	

SCHEDULED COMPLETION

ORIGINAL: NOT DETERMINED

CURRENT: N/S

• PROBLEM DESCRIPTION

Task A-48 was approved as a USI by the NRC in December 1980.

Postulated reactor accidents which result in a degraded or melted core can result in generation and release to the containment of large quantities of hydrogen. The hydrogen is formed from the reaction of the zirconium fuel cladding with steam at high temperatures and/or by radiolysis of water. Experience gained from the TMI-2 accident indicates that we may want to require more specific design provisions for handling larger hydrogen releases than currently required by the regulations particularly for smaller, low pressure containment designs.

This USI will investigate means to predict the quantity and release rate of hydrogen following degraded core accidents and various means to cope with large releases to the containment such as inerting of the containment or controlled burning. The potential effects of proposed hydrogen control measures on safety including the effects of hydrogen burns on safety related equipment will be investigated.

• RES INTERFACE INFORMATION

TO BE DEVELOPED

• ACRS INTERFACE INFORMATION

TO BE DEVELOPED

• TECHNICAL ASSISTANCE CONTRACTS

TO BE DEVELOPED

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED

• POTENTIAL PROBLEMS

• STATUS SUMMARY

A draft Task Action Plan (TAP) was issued for NRC comments on 4/02/82. A revised Draft Task Action Plan was issued for NRC comment on 11/03/82. A detailed schedule is tentatively scheduled to be issued by 01/31/83.

# INDICATES A CHANGE IN INFORMATION

PRESSURIZED THERMAL SHOCK (A-49)

AS OF WEEK ENDING NOVEMBER 19, 1982

KEY PERSONNEL

TASK MANAGER
ROY WOODS X24714

Roy Woods

MPA ANALYST
#B. ST. MARY X28330

TASK REVIEWERS

NAME	BRANCH
E. THROM	RSB/NRR
C. JOHNSON	RES
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J. STROSNIDER	RES
M. VAGINS	RES
R. JOHNSON	GIB/DST
R. KLECKER	MTEB/DE
N. RANDALL	MTEB/DE
G. VISSING	ORB-4/DL
J. CLIFFORD	PTRB/HFS

SCHEDULED COMPLETION

ORIGINAL: Not Determined

CURRENT: Not Scheduled

• PROBLEM DESCRIPTION

This task was designated a USI by the NRC in December 1981.

Neutron irradiation of reactor pressure vessel weld and plate materials decreases the fracture toughness of the materials. The fracture toughness sensitivity to radiation induced change is increased by presence of certain materials such as copper. Decreased fracture toughness makes it more likely that, if a severe overcooling event occurs followed by or concurrent with high vessel pressure, and if a small crack is present on the vessels inner surface, that crack could grow to a size that might threaten vessel integrity.

Severe pressurized overcooling events are improbable since they require multiple failures and improper operator performance. However, certain precursor events have happened that could have potentially threatened vessel integrity if additional failures had occurred and/or if the vessel had been more highly irradiated. Therefore, the possibility of vessel failure due to a severe pressurized overcooling event cannot be ruled out.

• RES INTERFACE INFORMATION

A major portion of the work is being performed under a contract with Oak Ridge National Laboratory through the Division of Risk Analysis, RES (FIN # B0468).

Other major contributors are:

Primary System Integrity Research Program through the Division of Engineering Technology, RES and

Code Applications Program through the Division of Accident Evaluation, RES

• ACRS INTERFACE INFORMATION

Meetings have been held and will be scheduled as necessary with the Subcommittee on Metallic Components and with the full ACRS. The next scheduled meetings are September 8 and 9, 1982, respectively.

• TECHNICAL ASSISTANCE CONTRACTS

Contract (B-2510) issued to PNL. PNL will provide their recommendations regarding the necessity of corrective actions, and will provide peer review of recommendations being considered by the NRC staff.

The following RES contracts are providing technical assistance to the PTS program. These are in addition to the technical assistance contracts which were initiated to specifically address the PTS issue and listed in the table below.

FIN No.	Description
B0119	HSST
B8133	LWR Pressure Boundary Integrity
B5988	Surveillance Dosimetry
B0415	Pressure Vessel Simulation
B6224	Dosimetry Meas. Data Base
B7026	JR Curve
A3215	Code Assessment and Application
A6047	Code Assessment and Applications
A7027	Analytical Res. in LWR Safety
A7217	TRAC Calc Assistance
B0468	Pressurized Thermal Shock

FIN NO.	CONTRACTOR	OBLIGATED	EXPENDED
#B-2510	PNL	$400K	$375K-est.
#A-7272	LANL	$210K-FY82 $210K-FY83	$ *
#A-3387	BNL	$210K	$ *

* Figures not available at the time of this publication.

• POTENTIAL PROBLEMS

As a result of the short-term findings, the A-49 network is going to be extensively revised.

• STATUS SUMMARY

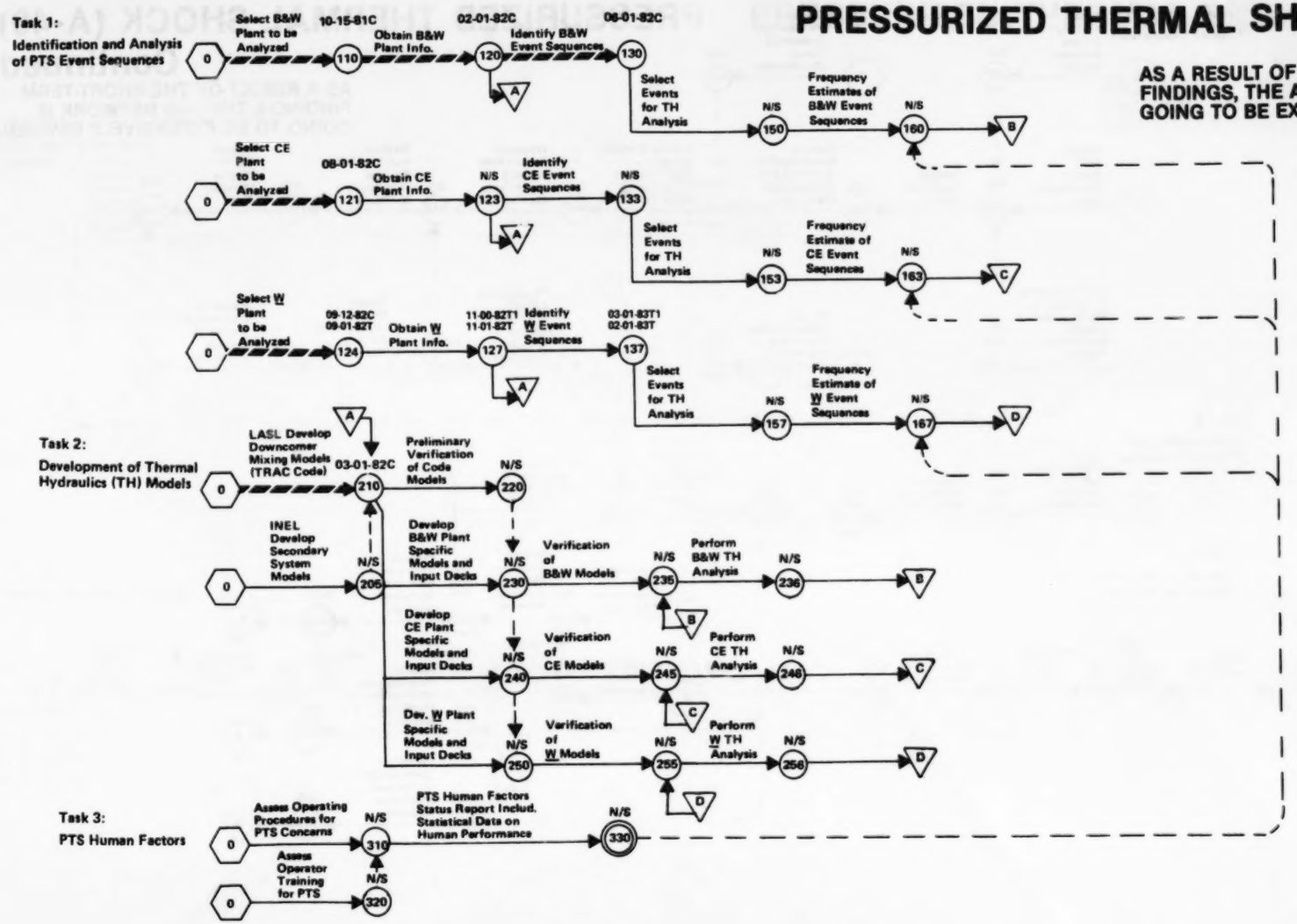
Draft short-term position is under preparation and will be completed in November 1982. Final PNL proposed position has been published as NUREG/CR-2837. ORNL has proposed the first three event sequences to be analyzed and they have been accepted by the staff. Review of the "150-day responses" has been completed. Results of the review are being integrated into the November staff position report.

The final Task Action Plan for A-49 was approved and issued on March 26, 1982.

# INDICATES A CHANGE IN INFORMATION

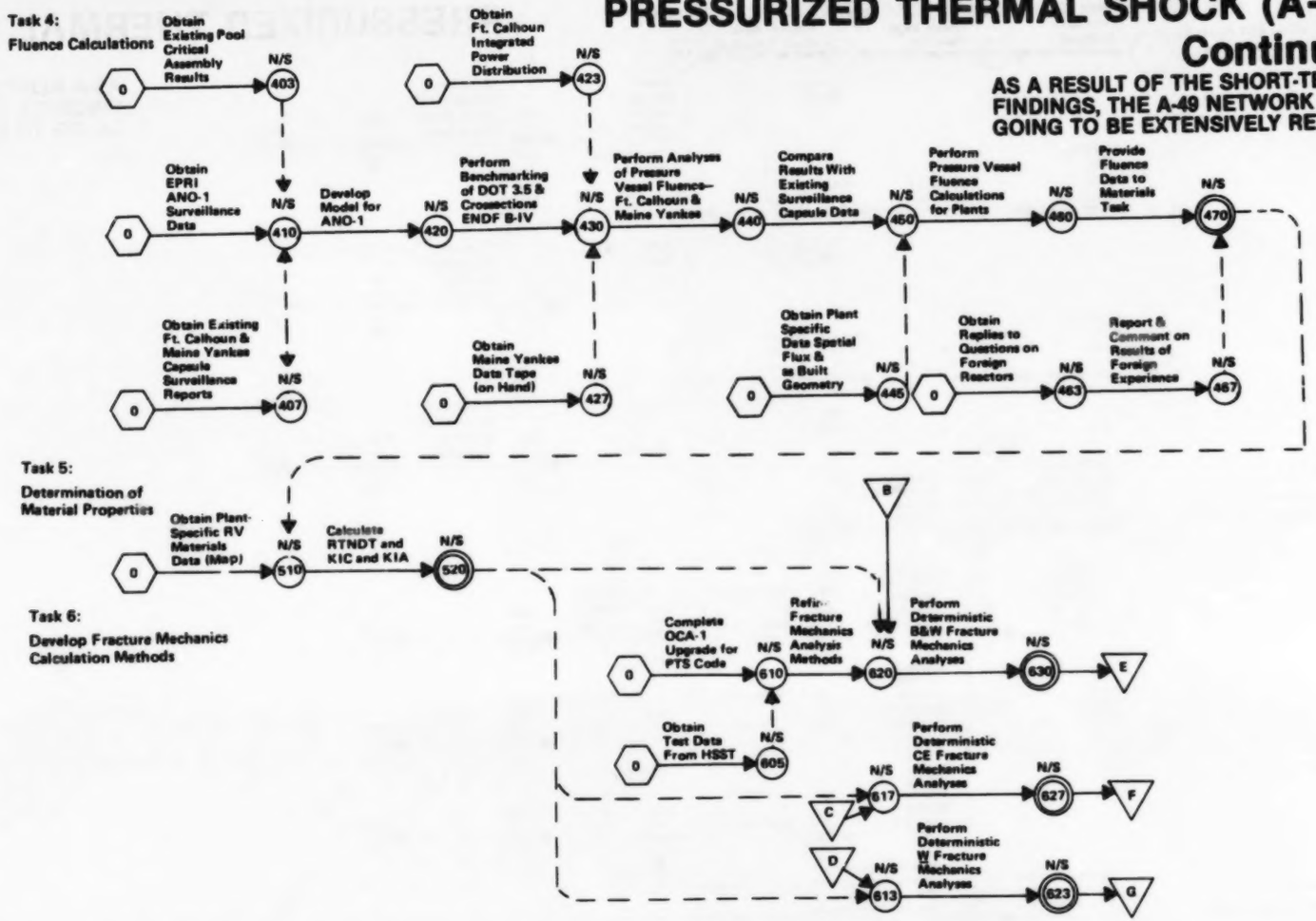
PRESSURIZED THERMAL SHOCK (A-49)

AS A RESULT OF THE SHORT-TERM FINDINGS, THE A-49 NETWORK IS GOING TO BE EXTENSIVELY REVISED.



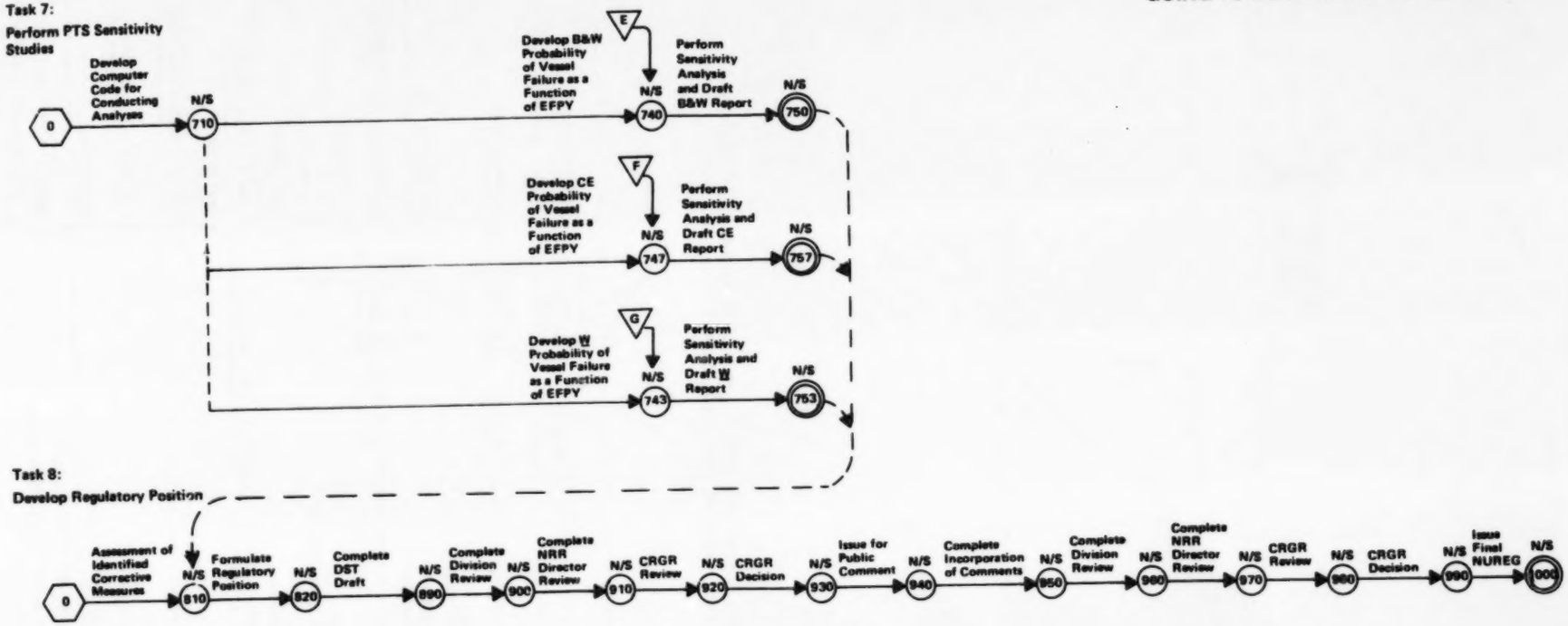
PRESSURIZED THERMAL SHOCK (A-49) Continued

AS A RESULT OF THE SHORT-TERM FINDINGS, THE A-49 NETWORK IS GOING TO BE EXTENSIVELY REVISED.



PRESSURIZED THERMAL SHOCK (A-49) Continued

AS A RESULT OF THE SHORT-TERM FINDINGS, THE A-49 NETWORK IS GOING TO BE EXTENSIVELY REVISED.



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